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10 CFR 50.90
10 CFR 50.69

March 11, 2021

U.S. Nuclear Regulatory Commission
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ATTN: Document Control Desk

Limerick Generating Station, Units 1 and 2
Renewed Facility Operating License Nos. NPF-39 and NPF-85
NRC Docket Nos. 50-352 and 50-353

Subject: Application to Implement an Alternate Defense-in-Depth Categorization Process, an Alternate Pressure Boundary Categorization Process, and an Alternate Seismic Tier 1 Categorization Process in Accordance with the Requirements of 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors"

References:

1. Limerick Generating Station, Units 1 and 2 – Issuance of Amendment Nos. 230 and 193 to Adopt Title 10 of the Code of Federal Regulations Section 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors" (CAC NOS. MF9873 and MF9874; EPID L-2017-LLA-0275)," July 31, 2018 (ADAMS Accession No. ML18165A162).

In accordance with the provisions of 10 CFR 50.69, and 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," Exelon Generation Company, LLC (Exelon) is requesting a revision to the license condition in Appendix C in the Renewed Facility Operating License Nos. NPF-39 and NPF-85 for Limerick Generating Station (Limerick), Units 1 and 2, respectively.

The NRC issued Amendments Nos. 230/193 and the Safety Evaluation for Limerick Units 1 and 2, respectively, to implement 10 CFR 50.69 in Reference 1. The amendments added a new license condition to the Renewed Facility Operating Licenses to allow the implementation of risk-informed categorization and treatment of structures, systems and components for Limerick in accordance with Title 10 of the Code of Federal Regulations Section 50.69.

The proposed amendments would modify the licensing basis by revising the license condition in Appendix C to allow the use of an alternate defense-in-depth categorization process, an alternate pressure boundary categorization process, and an alternate Seismic Tier 1 categorization process.

Enclosure 1 contains the evaluation of the proposed change. Enclosure 2 contains the markup of the license condition in Appendix C.

The alternate defense-in-depth categorization process, the alternate pressure boundary categorization process, and the alternate Seismic Tier 1 categorization process may be implemented for any system that was previously categorized, or systems that will be categorized. However, any system that has been previously categorized is not required to be re-categorized with the alternate defense-in-depth categorization process, the alternate pressure boundary categorization process, or the alternate Seismic Tier 1 categorization process. The categorization processes identified in the current license condition may continue to be used.

Limerick is the pilot plant for the industry's 10 CFR 50.69 alternate defense-in-depth and alternate pressure boundary categorization processes.

The proposed change has been reviewed by the Limerick Plant Operations Review Committee in accordance with the requirements of the Exelon Quality Assurance Program.

Exelon requests approval of the proposed license amendments by March 11, 2022, with the amendments being implemented within 60 days.

In accordance with 10 CFR 50.91, "Notice for public comment; State consultation," paragraph (b), Exelon is notifying the Commonwealth of Pennsylvania of this application for license amendments by transmitting a copy of this letter and its enclosures to the designated State Official.

This letter contains no regulatory commitments.

If you should have any questions regarding this submittal, please contact Glenn Stewart at 610-765-5529.

I declare under penalty of perjury that the foregoing is true and correct. Executed on this 11th day of March 2021.

Respectfully,

Rafferty-Czincila, Shannon

Digitally signed by Rafferty-Czincila, Shannon
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Shannon Rafferty-Czincila
Director - Licensing and Regulatory Affairs
Exelon Generation Company, LLC

Enclosures:
(1) Evaluation of the Proposed Change
(2) Proposed License Condition Mark-ups

cc:	USNRC Region I, Regional Administrator	w/ attachments
	USNRC Project Manager, LGS	"
	USNRC Senior Resident Inspector, LGS	"
	Director, Bureau of Radiation Protection - Pennsylvania Department of Environmental Protection	"

ENCLOSURE 1

License Amendment Request

Limerick Generating Station, Units 1 and 2
Docket Nos. 50-352 and 50-353

Application to Implement an Alternate Defense-in-Depth Categorization Process, an Alternate Pressure Boundary Categorization Process, and an Alternate Seismic Tier 1 Categorization Process in Accordance with the Requirements of 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors"

Evaluation of the Proposed Change

EVALUATION OF THE PROPOSED CHANGE

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1 SUMMARY DESCRIPTION

The Nuclear Regulatory Commission (NRC) issued Amendment Nos. 230 and 193 for Limerick Generating Station (Limerick), Units 1 and 2, respectively, to adopt Title 10 of the Code of Federal Regulations Section 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors" (CAC NOS. MF9873 and MF9874; EPID L-2017-LLA-0275)," dated July 31, 2018 (ADAMS Accession No. ML18165A162) (Reference [1]).

The amendments added a new license condition to the Renewed Facility Operating Licenses to allow the implementation of risk-informed categorization and treatment of structures, systems, and components for Limerick in accordance with the requirements of 10 CFR 50.69.

The proposed amendments would modify the licensing basis by revising the license condition in Appendix C to allow the use of an alternate defense-in-depth categorization process, an alternate pressure boundary categorization process, and an alternate Seismic Tier 1 categorization process. The processes are further discussed in the following industry documents:

1. PWROG-20015-NP, Revision 0, "Alternate 10 CFR 50.69 Defense-in-Depth Categorization Process," March 2021 (Reference [2]).

2. Electric Power Research Institute (EPRI) 3002015999, "Enhanced Risk-Informed Categorization Methodology for Pressure Boundary Components," November 2019 (Reference [3]).

3. EPRI 3002017583, "Alternative Approaches for Addressing Seismic Risk in 10 CFR 50.69 Risk-Informed Categorization," February 2020 (Reference [4]).

2 DETAILED DESCRIPTION

2.1 REASON FOR PROPOSED CHANGE

During the implementation of 10 CFR 50.69 by various licensees, it was determined that several processes are overly conservative when performing the 10 CFR 50.69 categorization and are resource intensive, without providing a commensurate benefit to the health and safety of the public. For example, when evaluating core damage defense-in-depth, credit cannot be taken for multiple identical, redundant trains. To address this, an alternate approach has been developed in lieu of the current defense-in-depth categorization process, the current pressure boundary categorization process (previously referred to as passive categorization as discussed in Section 3.5.4 of the Safety Evaluation (SE) that was issued for Limerick, Units 1 and 2, to implement 10 CFR 50.69 (Reference [1])), and the current seismic categorization process. The alternate defense-in-depth categorization process, the alternate pressure boundary categorization process, and the alternate Seismic Tier 1 categorization process are in compliance with 10 CFR 50.69; however, these processes allow additional focus on Risk-Informed Safety Class (RISC) RISC-1 and RISC-2 structures, systems and components (SSCs). The use of the alternate defense-in-depth categorization process, the alternate pressure boundary categorization process, and the alternate Seismic Tier 1 categorization process improves consistency and

removes subjectivity while reducing the 10 CFR 50.69 implementation effort to categorize systems.

Each of the alternate processes are briefly discussed below:

1. Alternate Defense-in-Depth Categorization Process

   a. The current core damage defense-in-depth process does not reflect the rigor of the current Probabilistic Risk Assessment (PRA) models and requires significant resources to evaluate each function/SSC using the initiating event frequency and success criteria obtained from the PRA model along with qualitative analysis. The alternate process is discussed in PWROG-20015-NP and uses the PRA model structure and insights to identify candidate High Safety Significant (HSS) functions/SSCs.

   b. The current containment defense-in-depth process does not reflect the rigor of today's PRA models and requires significant resources to evaluate each SSC using aspects of the PRA model and qualitative considerations. The alternate process uses the PRA model structure and insights to identify candidate HSS SSCs. The long-term containment integrity qualitative consideration in Section 6.2 of NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline," (Reference [5]) will continue to be evaluated.

2. Alternate Pressure Boundary Categorization Process

   a. The categorization of pressure boundary components does not efficiently use the PRA model and requires a significant effort to evaluate piping segments. This can be done more efficiently by using the PRA model and qualitative considerations to identify candidate HSS SSCs for a single plant level analysis rather than for each system individually.

3. Alternate Seismic Tier 1 Categorization Process

   a. The current seismic risk assessment method is a Seismic Margins Assessment (SMA). All SSCs included on the SMA Safe Shutdown Equipment List (SSEL), i.e., Success Path Component List (SPCL), conservatively default to HSS. The alternate Seismic Tier 1 categorization process employs a systematic process to evaluate the seismic hazard which is integrated into the categorization process. It considers likelihood and magnitude of the seismic hazard and margin to the site-specific design basis. The alternate Seismic Tier 1 categorization process is described in EPRI 3002017583.

   b. EPRI 3002017583 is an update to EPRI 3002012988, "Alternative Approaches for Addressing Seismic Risk in 10 CFR 50.69 Risk-Informed Categorization," July 2018 (Reference [6]) which was referenced in the NRC issued amendment and SE for Calvert Cliffs Nuclear Power Plant, Units 1 and 2, to implement 10 CFR 50.69 as noted below:

      Calvert Cliffs Nuclear Power Plant, Units 1 and 2, "Issuance of Amendment Nos. 332 and 310 Re: Risk-Informed Categorization and Treatment of Systems,

Structures, and Components (EPID L-2018-LLA-0482)," February 28, 2020. (ADAMS Accession No. ML19330D909) (Reference [7]).

This license amendment request incorporates by reference the Clinton Power Station, Unit 1 response to request for additional information letter of November 24, 2020 (ML20329A433) (Reference [8]), in particular, the response to the question regarding the differences between the initial EPRI 3002012988 and the current EPRI 3002017583 as well as Exelon's proposed approach for the 50.69 Seismic Alternative Tier 1.

2.2 DESCRIPTION OF THE PROPOSED CHANGE

The license condition in Appendix C currently states:

"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 SSCs and their associated supports; and the results of non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards, i.e., seismic margin analysis (SMA) to evaluate seismic risk, and a screening of other external hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009; as specified in Unit [1] License Amendment No. [230] dated July 31, 2018.

Exelon will complete the implementation items listed in Attachment 2 of Exelon letter to NRC dated April 23, 2018 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)."

The license condition in Appendix C is proposed to be revised as follows:

"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 SSCs and their associated supports; and the results of non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards, i.e., seismic margin

analysis (SMA) to evaluate seismic risk, and a screening of other external hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009; as specified in Unit [1] License Amendment No. [230] dated July 31, 2018.

In addition, Exelon is approved to implement 10 CFR 50.69 using any of the following alternate processes for categorization of RISC-1, RISC-2, RISC-3, and RISC-4 SSCs: the defense-in-depth approach contained in PWROG-20015-NP; the passive pressure boundary categorization approach described in EPRI 3002015999; and the seismic approach as described in Exelon's submittal letter dated March 11, 2021, as specified in Unit [1] License Amendment No. [XXX] dated [DATE].

Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from a seismic margins approach to a seismic probabilistic risk assessment approach)."

Note: The implementation items listed in Attachment 2 of Exelon letter to NRC dated April 23, 2018 were completed as required by the original license condition prior to the implementation of the 10 CFR 50.69 categorization process at Limerick which began in October 2018. Therefore, the paragraph specific to the implementation items is no longer applicable and is proposed to be deleted from the revised license condition for this license amendment request and replaced with the new insert paragraph for the alternate categorization processes as indicated above and in the proposed license condition markups in Enclosure 2.

3 TECHNICAL EVALUATION

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

3.1.1 Overall Categorization Process

The alternate defense-in-depth categorization process, the alternate pressure boundary categorization process, and the alternate Seismic Tier 1 categorization process are discussed in the following sections.

The alternate defense-in-depth categorization process, the alternate pressure boundary categorization process, and/or the alternate Seismic Tier 1 categorization process may be implemented for any system that was previously categorized or systems that will be categorized. However, any system that has been previously categorized is not required to be re-categorized with the alternate defense-in-depth categorization process, the alternate pressure boundary categorization process, or the alternate Seismic Tier 1 categorization process. The processes identified in the current License Condition Appendix C may continue to be used.

No assignments of Risk-Informed Safety Class (RISC) to SSCs are completed until a system is individually categorized, since categorization must be performed for entire systems and structures, not for selected components within a system or structure.

3.1.2 Alternate Defense-in-Depth Categorization Process

The alternate defense-in-depth categorization process discussed in PWROG-20015-NP, "Alternate 10 CFR 50.69 Defense-in-Depth Categorization Process," (Reference [2]) is proposed as an alternate to the guidance in NEI 00-04, Section 6. The assessment of long-term containment integrity in NEI 00-04 Section 6.2 will continue to be used (i.e., the current guidance is retained). The alternate defense-in-depth categorization process was piloted for ten Limerick systems that were previously categorized by current methods and the results were compared. The systems selected represent both front line and support systems which adequately exercise the pilot process. The alternate defense-in-depth process was developed to consider the improved PRA modeling that has evolved since NEI 00-04 was first issued in 2005, and the categorization experience gained in more recent actual implementation of 10 CFR 50.69 at Limerick. Use of the PRA model logic structure presents a more effective and consistent method of assessing defense-in-depth for 10 CFR 50.69 categorization. The use of the model logic assures that key safety functions are still maintained by redundant SSCs, and better addresses the qualitative considerations identified in Section 9.2.2 of NEI 00-04. Although the PWROG alternate categorization process does not change the Integrated Decision-making Panel (IDP) defense-in-depth assessment described in Section 9 of NEI 00-04, this revised approach, that utilizes the PRA model, was found to be more objective than the set of qualitative considerations discussed in that Section.

The revised screening approach proposed as an alternate to the guidance in NEI 00-04, Section 6.1, uses the Full Power Internal Events (FPIE) PRA Model. The proposed approach identifies cutsets that have an initiating event and a single basic event representing a failure of an SSC, including an independent failure, common cause failure, or a human failure event which leads to core damage. Cutsets with initiating event frequencies that are less than 1E-04/yr are screened out since NEI 00-04, Figure 6-1 has low safety significance confirmed with a frequency less than 1E-03/yr. In the filtered cutsets, the SSCs are identified for the initiating events and basic events. The SSCs from the filtered cutsets are considered candidate HSS and the associated functions are driven to candidate HSS following the process in NEI 00-04, Section 7.1. Per PWROG-20015-NP, pressure boundary failure initiating events and pressure boundary failure basic events are not addressed by this Core Damage Defense-in-Depth alternate categorization process (e.g., pipe ruptures leading to internal flooding scenarios). The same process used for Core Damage Defense-in-Depth described above is followed for Containment Defense-in-Depth with the exceptions that the FPIE LERF PRA model is utilized and each system categorized continues using the NEI 00-04, Section 6.2, Long-Term Containment Integrity guidance.

Several prerequisites outlined in the PWROG approach must be met in order to implement the enhanced alternate categorization process. PWROG-20015-NP stipulates that the FPIE PRA model meet the following Standard Supporting Requirements. The Limerick FPIE PRA model meets the prerequisites as described below.

1. The PRA model used for the defense-in-depth evaluations is acceptable for implementation of 50.69 by the NRC. The NRC issued its safety evaluation for the Limerick §50.69 LAR on July 31, 2018 (Reference [1]).
2. Findings related to the accident sequence analysis must be closed or dispositioned as not impacting the defense-in-depth alternate categorization process. There are no open

findings associated with accident sequence analysis in the Limerick PRA model that impact the alternate approach to the defense-in-depth assessment.

3. Findings related to the success criteria must be closed or dispositioned as not impacting the defense-in-depth alternate categorization process. There are no open findings associated with success criteria in the Limerick PRA model that impact the alternate approach to the defense-in-depth assessment.
4. Findings related to the initiating event frequencies must be closed or dispositioned as not impacting the defense-in-depth alternate categorization process. There are no open findings associated with initiating event frequencies in the Limerick PRA model that impact the alternate approach to the defense-in-depth assessment.
5. Findings related to truncation must be closed or dispositioned as not impacting the defense-in-depth alternate categorization process. There are no open findings associated with truncation in the Limerick PRA model that impact the alternate approach to the defense-in-depth assessment.
6. Findings related to common cause groupings must be closed or dispositioned as not impacting the defense-in-depth alternate categorization process. There are no open findings associated with common cause groupings in the Limerick PRA model that impact the alternate approach to the defense-in-depth assessment.

The results for the ten Limerick systems that were piloted more accurately reflect the as-built as-operated plant by making use of the FPIE PRA and Level 2 LERF models. The SSCs that screened as HSS were reasonable in that they and their associated functions are relied upon in multiple accident mitigation strategies without adequate redundancy and diversity. Logically, SSCs and functions that are modeled in the FPIE PRA for events that could cause core damage or impact LERF should produce similar results in the approach presented in NEI 00-04 for the PRA assessments, Section 5, and the Defense-in-depth assessments, Section 6. The pilot resulted in fewer HSS functions from defense-in-depth and fewer HSS components overall. While the results show a decrease in the number of HSS components, the process ensures that adequate redundancy and diversity is identified and retained for Low Safety Significant (LSS) components. The alternate defense-in-depth approach removes the subjectivity of the NEI-00-04, Section 6 considerations when applied to non-front line accident mitigation systems. Of the ten systems that were piloted, five that had functions considered HSS from the original NEI 00-04, Section 6 defense-in-depth approach, now were considered LSS by the alternate defense-in-depth approach. Each of these five systems’ key safety functions were determined to be maintained by redundant SSCs. The pilot results were found to be reasonable and consistent, indicating that the current NEI 00-04 Section 6 approach was/is conservative.

Candidate HSS SSCs from the alternate defense-in-depth process are developed using the FPIE PRA model. This candidate list provides the input to the final defense-in-depth categorization for any system selected for categorization under 10 CFR 50.69.

The alternate defense-in-depth categorization process meets the guidance for defense-in-depth in Regulatory Guide 1.174 as discussed in PWROG-20015-NP.

3.1.3 Alternate Pressure Boundary Categorization Process

When implementing the 10 CFR 50.69 categorization process, pressure boundary components are those components that perform a pressure retaining function. This was previously referred to as passive categorization in Section 3.5.4 of the NRC Safety Evaluation that was issued for Limerick Units 1 and 2 to implement 10 CFR 50.69.

The pressure boundary categorization process described in EPRI 3002015999, "Enhanced Risk-Informed Categorization Methodology for Pressure Boundary Components," is proposed as an alternate to the ANO-2 R&R-004 (Reference [9]) methodology found in the current Limerick license condition. The EPRI methodology stipulates that the plant have a technically adequate PRA including an internal flooding PRA. The Limerick internal events and plant flooding PRA is of sufficient technical adequacy for use in this enhanced approach for pressure boundary categorization. The PRA model was previously shown to be sufficient to support 50.69 (Reference [1]) and substantial experience has been gained by the subsequent categorization of over 20 Limerick plant systems. Limerick also has a risk-informed ISI (RI-ISI) program that relies on insights from this PRA model. Several additional prerequisites outlined in the EPRI study must also be met, in order to implement the enhanced methodology. These prerequisites are met for Limerick as follows:

1. Limerick has a robust internal events and flooding plant PRA. EPRI 1021467, "Nondestructive Evaluation: Probabilistic Risk Assessment Technical Adequacy Guidance for Risk-Informed In-Service Inspection Programs," (Reference [10]) contains guidance and scope sufficient for use in 10 CFR 50.69 categorization of pressure retaining components. The Limerick RI-ISI program has been in existence for almost 20 years. Lastly, the internal flooding model is robust and of sufficient quality to support the EPRI methodology (reference Section 3.2 below). Internal flooding protective measures (e.g., floor drains, sumps, flood alarms, and barriers) will be considered LSS, unless additional evaluations show that failure of these protective measures will invalidate LSS determinations of pressure retaining components. Therefore, Limerick meets this prerequisite.

2. Limerick has a robust program that addresses localized corrosion (e.g., pitting and microbiologically influenced corrosion). The program follows the guidance in the following EPRI technical reports:
    - TR-103403 (service water corrosion) (Reference [11]),
    - 3002003190 (service water chemical addition systems) (Reference [12]),
    - TR-102063 (examination of service water systems) (Reference [13]),
    - 1010059 (service water piping guidelines) (Reference [14]), and
    - 3002018352 (1016456, Revision 2 which is an update to 1016456, Revision 0 referenced in EPRI 300201599) (management of buried piping) (Reference [15]).

    Therefore, Limerick meets this prerequisite.

3. The Exelon Flow-Accelerated Corrosion (FAC) Program follows the guidance in the industry standard document, EPRI 3002000563 (NSAC 202L R4, Recommendations for an Effective Flow-Accelerated Corrosion Program) (Reference [16]). Additionally, the FAC programs implement the use of standardized health reporting that is consistent with

those developed out of NEI Efficiency Bulletin 16-34, "Streamline Program Health Reporting." Therefore, Limerick meets this prerequisite.

4. The Exelon Erosion in Piping and Components (EPC) Program follows the guidance in the industry standard document EPRI 3002005530 (Recommendations for an Effective Program Against Erosive Attack) (Reference [17]) of which Exelon was a contributor towards development. Therefore, Limerick meets this prerequisite.

The enhanced pressure boundary methodology does not represent a fundamental change in process, but rather improves upon the guidance used to perform pressure boundary evaluations. The improved methodology represents a potential for a more efficient process since it is performed for all piping segments in the plant (not on a system by system basis) regardless of whether those systems are subjected to full categorization. In addition, the following specific guidance and additional clarifications are added for a more consistent evaluation:

1. Eliminates redundancy in certain qualitative steps captured by other considerations (e.g., impacts on shutdown).
2. Clarifies that ASME Class 1 Exempt piping is LSS since ruptures in these lines do not exceed normal makeup capability.
3. Clarifies that supports need not be categorized until such time as the need is identified.
4. Improves guidance for passive SSCs predetermined as HSS, such as heat exchangers whose failure could allow reactor coolant to bypass containment, e.g., residual heat removal heat exchanger internal tube failures that could add contaminated fluid to the ultimate heat sink (UHS) or that could impact multiple systems.
5. Redefines pressurized water reactor (PWR) interfacing-systems loss-of-coolant accident (ISLOCA) sections of interest and clarifies which unisolable leaks that provide inventory to multiple systems are HSS (i.e., sources up to the first isolation valve).
6. Provides a more precise HSS definition for portions of the UHS flow path (e.g., service water piping whose failure will fail both trains) resulting in the failure of the UHS function.
7. Adds a new HSS sliding scale criterion for ruptured piping/components (modeled in the PRA) that contribute > 1E-08/year and the product of CDF contribution times CCDP > 1E-08/year.
8. Adds new internal flooding criterion for segments that contribute 1E-6 CDF (1E-7 LERF).
9. Better defines the scope for predetermined HSS components within the break exclusion region (BER) for high energy piping segments (high energy line break) piping systems outside containment, i.e., larger than Nominal Pipe Size (NPS) 4.

The EPRI pressure boundary methodology was piloted at Limerick in April 2020. Consistent with the methodology described in EPRI 3002015999, the methodology was applied to the entire plant. The results were found to be reasonable and consistent, indicating that the current ANO-2 R&R-004 methodology is conservative. The most noticeable change in the results of the categorization between the pilot and the existing 50.69 process is the change in categorization to LSS of the Class 1 Exempt piping segments that were defaulted to HSS in the current process independent of consequence to the site. In general, these segments pose a low PRA consequence risk; therefore, categorizing them LSS in the EPRI methodology is

acceptable. Another notable change was the inclusion of all components within the BER. Although these piping sections would now be HSS for the systems categorized at Limerick under the current 50.69 process, there would not be a major impact on the number of HSS components as there are very few components in this piping region.

3.1.4 Alternate Seismic Tier 1 Categorization Process

The NRC previously issued its Safety Evaluation for Limerick approving the 10 CFR 50.69 process (Reference [1]).

10 CFR 50.69(c)(1) requires the use of PRA to assess risk from internal events. For other risk hazards such as seismic, 10 CFR 50.69(b)(2) allows, and NEI 00-04 (Reference [5]) 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 Limerick seismic hazard assessment, Exelon proposes to use a risk informed graded approach that meets the requirements of 10 CFR 50.69(b)(2) as an alternative to those listed in NEI 00-04 Sections 1.5 and 5.3. This approach is specified in EPRI 3002017583, "Alternative Approaches for Addressing Seismic Risk in 10 CFR 50.69 Risk-Informed Categorization," Reference [4], and includes additional qualitative considerations that are discussed in this section.

Limerick meets the EPRI 3002017583 Tier 1 criteria for a "Low Seismic Hazard/High Seismic Margin" site. The Tier 1 criteria are as follows:

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

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

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

> *"At Tier 1 sites, the likelihood of identifying a unique seismic condition that would cause an SSC to be designated HSS is very low.*

> *Therefore, with little to no anticipated unique seismic insights, the 50.69 categorization process using the FPIE PRA and other risk evaluations along with the required Defense-in-Depth and IDP qualitative considerations are expected to adequately identify the safety-significant functions and SSCs required for those functions and no additional seismic reviews are necessary for 10 CFR 50.69 categorization."*

The proposed categorization approach for Limerick 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 [4] along with confirmation that the site GMRS is low. Reference [4] demonstrates that seismic risk is adequately addressed for Tier 1 sites by the results of additional qualitative assessments discussed in this section and existing elements of the 10 CFR 50.69 categorization process specified in NEI 00-04.

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

EPRI 3002017583 recommends a risk-informed graded approach for addressing the seismic hazard in the 10 CFR 50.69 categorization process. There are a number of seismic fragility fundamental concepts that support a graded approach and there are important characteristics about the comparison of the seismic design basis (represented by the SSE) to the site-specific seismic hazard (represented by the GMRS) that support the selected thresholds between the three evaluation Tiers in the EPRI report. The coupling of these concepts with the categorization process in NEI 00-04 are the key elements of the approach defined in EPRI 3002017583 for identifying unique seismic insights.

The seismic fragility of an SSC is a function of the margin between an SSC's seismic capacity and the site-specific seismic demand. References such as EPRI NP-6041 (Reference [18]) 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 Limerick, there is no need to perform more detailed evaluations to demonstrate the inherent seismic capacities documented in industry sources such as Reference [18]. Low seismic demand sites have lower likelihood of seismically-induced failures and lesser challenges to plant systems. This, therefore, provides the technical basis for allowing use of a graded approach for addressing seismic hazard at Limerick.

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

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

a. SSCs for which the inherent seismic capacities are applicable, or which are designed to the plant SSE will have low probabilities of failure at sites where the peak spectral acceleration of the GMRS < 0.2g or where the GMRS < SSE between 1 and 10 Hz.
b. The low probabilities of failure of individual components would also apply to components considered to have correlated seismic failures.
c. These low probabilities of failure lead to low seismic CDF and LERF estimates, from an absolute risk perspective.
d. The low seismic CDF and LERF estimates lead to reasonable confidence that seismic risk contributions would allow reducing a HSS to LSS due to the 10 CFR 50.69 Integral Assessment if the equipment is HSS only due to seismic considerations.

Test cases described in Section 3 of Reference [4] 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 [4] test case information Exelon is using from other licensees and being incorporated by Reference into this application is described in Case Study A (Reference [19], [20], [21]), Case Study C (References [22], [23]), and Case Study D (References [24], [25], [26], [27], [28]). 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 Limerick, correlation studies would not lead to new seismic insights or affect the baseline seismic CDF in any significant way.

The Tier 1 to Tier 2 threshold as defined in EPRI 3002017583 provides a clear and traceable boundary that can be consistently applied plant site to plant site. Additionally, because the boundary is well defined, if new information is obtained on the site hazard, a site's location within a particular Tier can be readily confirmed. In the unlikely event that the Limerick seismic hazard changes to medium risk (i.e., Tier 2) at some future time, Exelon 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 Limerick meets the Tier 1 site criteria. In response to the NRC 50.54(f) letter associated with post-Fukushima recommendations (Reference [29]), Exelon submitted a seismic hazard screening report (Reference [30]) to the NRC. The GMRS for Limerick is below or approximately equal to the SSE between 1 Hz and 10 Hz and therefore meets the Tier 1 criterion in Reference [4].

The Limerick SSE and GMRS curves from the seismic hazard and screening response in Reference [30] are shown in Figure 1. The NRC's staff assessment of the Limerick seismic

hazard and screening response is documented in Reference [31]. In section 3.4 of Reference [31], 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 Limerick site.

Section 1.1.3 of Reference [4] cites various post-Fukushima seismic reviews performed for the U.S. fleet of nuclear power plants. For Limerick, 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 [30], [31]).
2. NTTF Recommendation 2.3 seismic walkdowns (References [32], [33], [34]).
3. NTTF Recommendation 4.2 seismic mitigation strategy assessment (S-MSA). (References [35], [36])

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

4. NTTF Recommendation 2.1 seismic high frequency evaluation (References [37], [38]).

As an enhancement to the EPRI study results as they pertain to Limerick, the proposed Limerick 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 Limerick. For example, as part of the categorization team's preparation of the System Categorization Document (SCD) that is presented to the IDP, a section will be included in the SCD that summarizes the identified plant seismic insights pertinent to the system being categorized, and will also state the basis for applicability of the EPRI 3002017583 study and the bases for Limerick 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 2 and Table 1 below) 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 Limerick) 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 Limerick 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 Limerick 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:

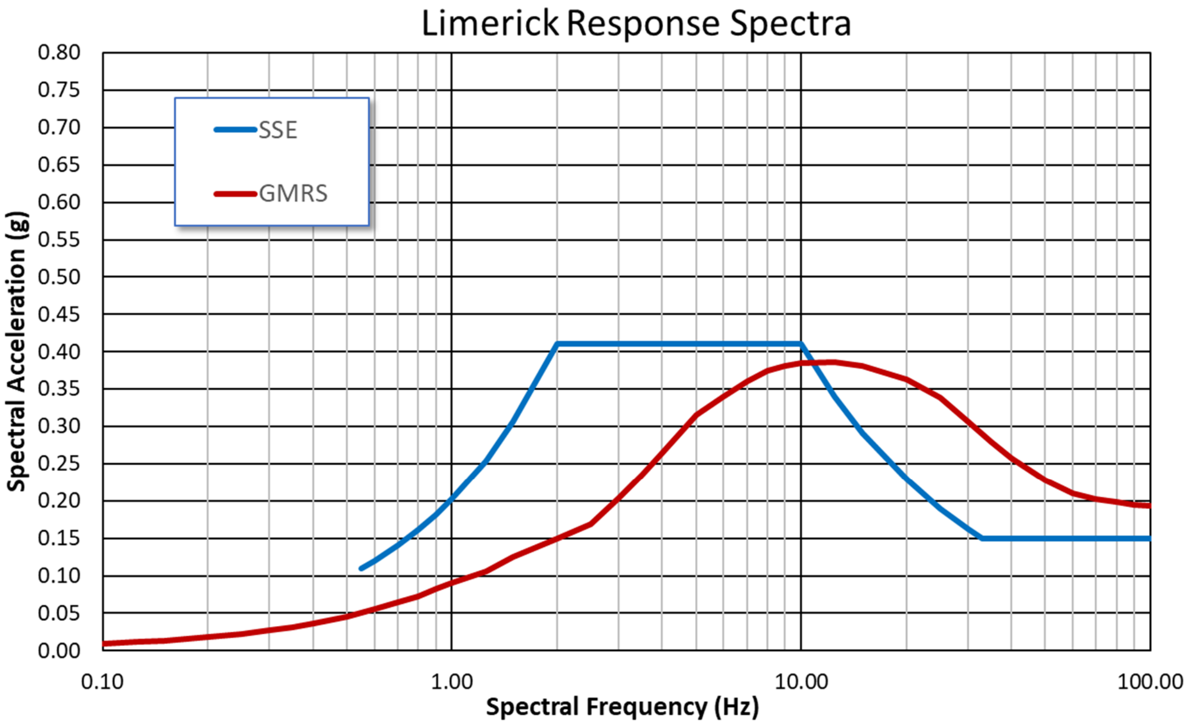
- Impact of relay chatter
- Implications related to potential seismic interactions such as with block walls
- Seismic failures of passive SSCs such as tanks and heat exchangers
- Any known structural or anchorage issues with a particular SSC
- Components that are implicitly part of PRA-modeled functions (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 2. As such, the IDP can challenge, from a seismic perspective, any candidate LSS recommendation for any SSC if they believe there is basis for doing so. Any decision by the IDP to downgrade preliminary HSS components to LSS will also consider the applicable seismic insights in that decision. These insights will provide the IDP a means to consider potential impacts of seismic events in the categorization process.

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

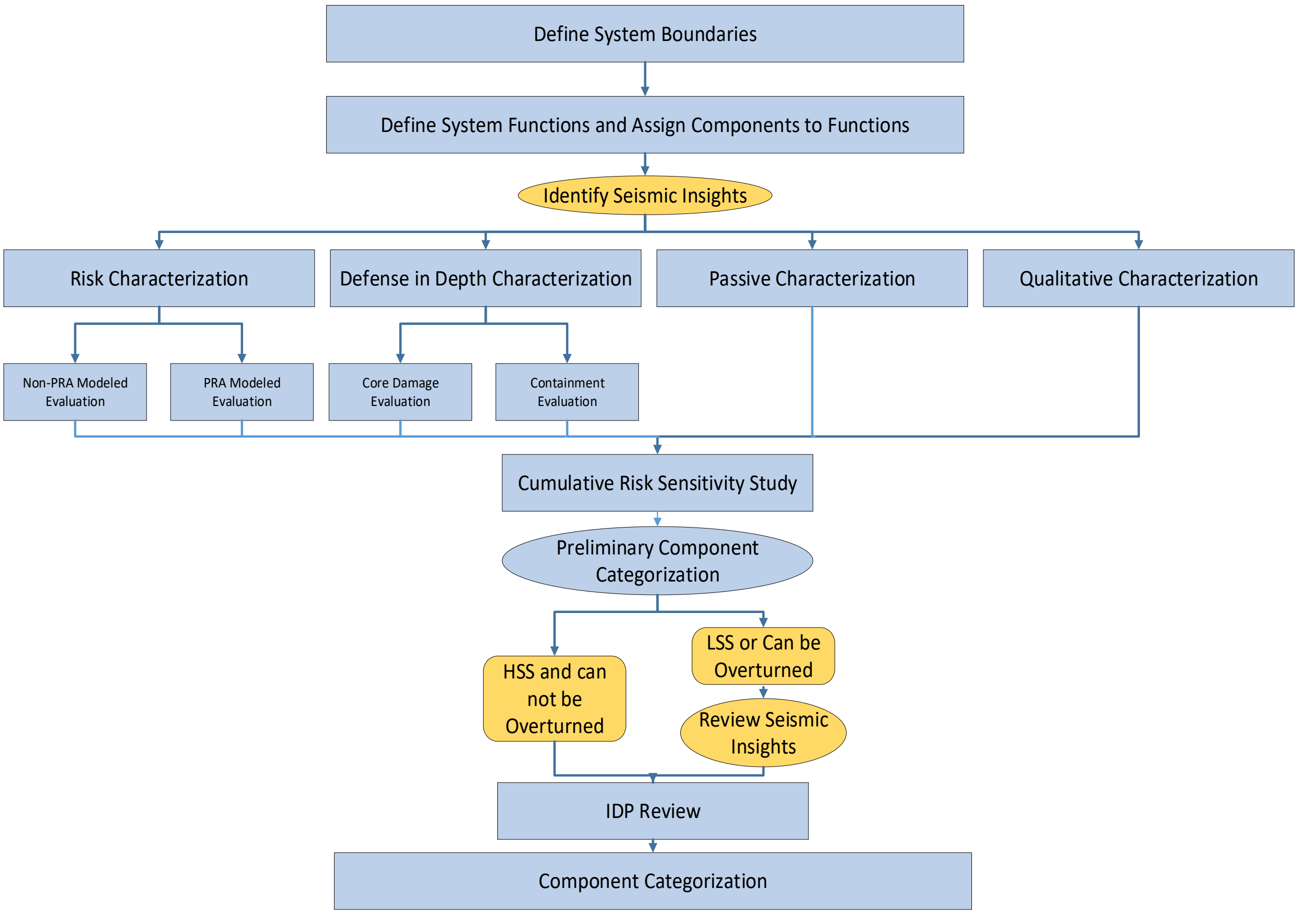
Based on the above, the Summary/Conclusion/Recommendation from Section 2.2.3 of Reference [4] applies to Limerick, i.e., Limerick is a Tier 1 plant for which the GMRS is very low such that unique seismic categorization insights are expected to be minimal. As discussed in Reference [4], the likelihood of identifying a unique seismic insight that would cause an SSC to be designated HSS is very low. References [39], [40], and [41] are incorporated into this LAR as they provide additional supporting bases for Tier 1 plants. Therefore, with little to no anticipated unique seismic insights, the 10 CFR 50.69 categorization process using the FPIE PRA and other risk evaluations along with the defense-in-depth and qualitative assessment by the IDP adequately identify the safety-significant functions and SSCs.

Figure 1
GMRS and SSE Response Spectra for Limerick[1]



[1] (From Reference [30], Table 2.4-1 (GMRS) and Table 3.1-1 (SSE))

Figure 2
Categorization Process Overview



**Note: The above figure is an update to Figure 5-1 in Reference [42].

Table 1: Categorization Evaluation Summary

(Table 1 below is an update to Table 1 in Reference [43]. Items in **BOLD** are new and applicable to Limerick.)

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
	EPRI Tier 1 Seismic	**Function/Component**	**Allowed (Note 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
	PWROG Alternate DID	**Function/Component**	**Not Allowed**	**Yes**
Qualitative Criteria	Considerations – Section 9.2	Function	Allowable (Note 1)	N/A
Passive	Passive – Section 4	Segment/Component	Not Allowed	No
	EPRI Enhanced Passive	**Segment/Component**	**Not Allowed**	**No**

<u>Notes</u>

[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 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.

3.2 TECHNICAL ADEQUACY EVALUATION (10 CFR 50.69(b)(2)(ii))

The full FPIE peer review was conducted in November 2005, and a focused-scope peer review for internal flood was conducted in May 2008. The 2005 FPIE peer review findings and the 2008 internal flood peer review findings were addressed and subsequently incorporated in the LGS PRA model. On July 17-22, 2016, an independent technical review of the findings and their resolution was performed by an independent Finding Closure Review team (Reference [44]). The purpose was to perform an independent assessment to review close out of "Finding" level F&Os of record from prior PRA peer reviews against the ASME/ANS PRA Standard. The independent assessment was observed at the time by the NRC.

Following that closure review, several findings related to the FPIE PRA model (including internal flood) remained open or partially resolved. Subsequently, the LGS FPIE PRA model was updated and the majority of the findings were addressed. In addition, a focused scope peer review was performed in August 2018 of a change considered an upgrade (Reference [45]). The dispositions of those remaining open FPIE PRA findings with respect to this application are provided in Appendix A below. Findings listed as ADDRESSED are those findings that have been addressed in the current PRA model but were not resolved by a finding closure review.

The technical adequacy was previously reviewed and approved by the NRC in the Safety Evaluation that was issued for Limerick Units 1 and 2 to implement 10 CFR 50.69 (Reference [1]). The technical adequacy evaluation is not impacted by these proposed changes.

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

Exelon may implement the alternate defense-in-depth categorization process, the alternate pressure boundary categorization process, and/or the alternate Seismic Tier 1 categorization process discussed in Section 3.1. The processes identified in the current license condition may continue to be used.

The overall risk evaluation process discussed in PWROG-20015-NP, EPRI 3002015999, EPRI 3002017583 addresses both known degradation mechanisms and common cause interactions and meets the requirements of 10 CFR 50.69(b)(2)(iv).

The sensitivity studies discussed in Section 8 of NEI 00-04, will be used to confirm that the categorization process results in acceptably small increases to CDF and LERF.

The SSC failure rates and initiating event frequencies used in the Limerick PRA include the quantifiable impacts from known degradation mechanisms, as well as other mechanisms (e.g., design errors, manufacturing deficiencies, human errors, etc.).

Subsequent performance monitoring and PRA updates as required by 10 CFR 50.69 will continue to include this data and provide timely insights into the need to account for any important new degradation mechanisms.

3.4 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 Seismic Tier 1 approach discussed in section 3.1.4 above, 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 assurance, 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.
- 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.
- NRC 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 alternate defense-in-depth categorization process, the alternate pressure boundary categorization process, and the alternate Seismic Tier 1 categorization process continue to meet the above regulation and regulatory guidance.

4.2 NO SIGNIFICANT HAZARDS CONSIDERATION ANALYSIS

In accordance with the provisions of 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors," and 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," Exelon Generation Company, LLC (Exelon) is requesting a revision to the license condition in Appendix C in the Renewed Facility Operating License Nos. NPF-39 and NPF-85 for Limerick Generating Station (Limerick), Units 1 and 2, respectively.

The NRC issued Amendments Nos. 230/193 and the Safety Evaluation for Limerick Units 1 and 2, respectively, to implement the requirements of 10 CFR 50.69 in Reference [1]. The amendments added a new license condition to the Renewed Facility Operating Licenses to allow the implementation of risk-informed categorization and treatment of structures, systems, and components for Limerick in accordance with Title 10 of the Code of Federal Regulations Section 50.69.

The proposed amendments would modify the licensing basis by revising the license condition in Appendix C to allow the use of an alternate defense-in-depth categorization process, an alternate pressure boundary categorization process, and an alternate Seismic Tier 1 categorization process.

Exelon has evaluated whether a significant hazards consideration is involved with the proposed amendments 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 the alternate defense-in-depth categorization process, the alternate pressure boundary categorization process, and the alternate Seismic Tier 1 categorization process for the 10 CFR 50.69 risk-informed categorization process to modify the scope of SSCs subject to NRC special treatment requirements and to implement alternative treatments per the regulations. The process used to evaluate structures, systems, and components (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 the alternate defense-in-depth categorization process, the alternate pressure boundary categorization process, and the alternate Seismic Tier 1 categorization process for the 10 CFR 50.69 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 the alternate defense-in-depth categorization process, the alternate pressure boundary categorization process, and the alternate Seismic Tier 1 categorization process for the 10 CFR 50.69 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 a 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 amendments 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 amendments 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 amendments do 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 amendments meet 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 amendments.

6 REFERENCES

[1] Limerick Generating Station, Units 1 And 2 - Issuance of Amendment Nos. 230 and 193 to Adopt Title 10 of the Code of Federal Regulations Section 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors," (CAC NOS. MF9873 AND MF9874; EPID L-2017-LLA-0275), July 31, 2018 (ML18165A162).

[2] PWROG-20015-NP, "Alternate 10 CFR 50.69 Defense-in-Depth Categorization Process," PA-RMSC-1769, Revision 0, March 2021.

[3] Enhanced Risk-Informed Categorization Methodology for Pressure Boundary Components, EPRI, Palo Alto, CA: November 2019. 3002015999.

[4] Alternative Approaches for Addressing Seismic Risk in 10 CFR 50.69 Risk-Informed Categorization, EPRI, Palo Alto, CA: February 2020. 3002017583.

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

[6] Alternative Approaches for Addressing Seismic Risk in 10 CFR 50.69 Risk-Informed Categorization, EPRI, Palo Alto, CA: July 2018. 3002012988.

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

[8] Clinton Power Station, Unit 1, "Response to Request for Additional Information Regarding License Amendment Requests to Adopt TSTF-505, Revision 2, and 10 CFR 50.69," November 24, 2020 (ADAMS Accession No. ML20329A433).

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

[10] Nondestructive Evaluation: Probabilistic Risk Assessment Technical Adequacy Guidance for Risk-Informed In-Service Inspection Programs, EPRI, Palo Alto, CA: 2011. 1021467.

[11] Service Water System Corrosion and Deposition Sourcebook, EPRI, Palo Alto, CA: 1993. TR-103403.

[12] Engineering and Design Considerations for Service Water Chemical Addition Systems, EPRI, Palo Alto, CA: 2014. 3002003190.

[13] Guide for the Examination of Service Water System Piping, EPRI, Palo Alto, CA: 1994. TR-102063.

[14] Service Water Piping Guideline, EPRI, Palo Alto, CA: 2005. 1010059.

[15] Recommendations for an Effective Program to Control the Degradation of Buried and Underground Piping and Tanks (1016456, Revision 2), EPRI, Palo Alto, CA: 2020. 3002018352.

[16] Recommendations for an Effective Flow-Accelerated Corrosion Program (NSAC-202L-R4), EPRI, Palo Alto, CA: 2013. 3002000563.

[17] Recommendations for an Effective Program Against Erosive Attack, EPRI, Palo Alto, CA: 2015. 3002005530.

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

[19] Peach Bottom Atomic Power Station, Units 2 and 3, Seismic Probabilistic Risk Assessment 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," dated August 28, 2018 (RS-18-098) (ML18240A065).

[20] Peach Bottom Atomic Power Station, Units 2 and 3 - Staff Review of Seismic Probabilistic Risk Assessment, "Associated with Reevaluated Seismic Hazard Implementation of the Near-Term Task Force Recommendation 2.1: Seismic," (EPID NO. L-2018-JLD-0010), June 10, 2019 (ML19053A469).

[21] Peach Bottom Atomic Power Station, Units 2 and 3 - Correction Regarding Staff Review of Seismic Probabilistic Risk Assessment, "Associated with Reevaluated Seismic Hazard Implementation of the Near-Term Task Force Recommendastion 2.1: Seismic," (EPID NO. L-2018-JLD-0010), October 8, 2019 (ML19248C756).

[22] Plant C Nuclear Plant, Units 1 and 2, License Amendment Request to Modify Approved 10 CFR 50.69 Categorization Process, June 22, 2017 (ML17173A875).

[23] Plant C Nuclear Plant, Units 1 and 2, "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).

[24] 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).

[25] Plant D Nuclear Plant, Units 1 and 2, Seismic Probabilistic Risk Assessment Supplemental Information, April 10, 2018 (ML18100A966).

[26] Plant D Nuclear Plant, Units 1 and 2 - Staff Review of Seismic Probabilistic Risk Assessment Associated With Reevaluated Seismic Hazard Implementation, of the NTTF Recommendation 2.1: Seismic (CAC NOS. MF9879 AND MF9880; EPID L-2017-JLD-0044) July 10, 2018 (ML18115A138).

[27] 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).

[28] Plant D Nuclear Plant, Units 1 And 2 - Issuance of Amendment Nos. 134 And 38 Regarding , Adoption of 10 CFR 50.69, "Risk-Informed Ategorization And Treatment Of Structures, Systems, And Components For Nuclear Power Plants" (EPID L-2018-LLA-0493) April 30, 2020 (ML20076A194).

[29] 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).

[30] Limerick Generating Station, Units 1 and 2, Exelon Generation Company, LLC, Seismic Hazard and Screening Report (Central and Eastern United States (CEUS) Sites), Response to NRC Request for Information Pursuant to 10 CFR 50.54(f), Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident, March 31, 2014 (ML14090A236).

[31] Limerick Generating Station, Units 1 and 2 - Staff Assessment of Information Provided Pursuant to 10 CFR 50.54(f), "Seismic Hazard Reevaluations for Recommendation 2.1 of the NTTF Review of Insights from the Fukushima Dai-Ichi Accident," (TAC NOS. MF3864 and MF3865), November 6, 2015 (ML15296A492).

[32] Limerick, Units 1 and 2, 180-Day Response to NRC Request for Information Pursuant to 10 CFR 50.54(f), "Seismic Aspects of Recommendation 2.3 of the NTTF Review of Insights from the Fukushima Dai-ichi Accident," Release Date December 17, 2012 (ADAMS Package ML123420141).

[33] Limerick Generating Station, Unit 1- Staff Assessment of the Seismic Walkdown Report Supporting Implementation of Near-Term Task Force Recommendation 2.3, Related to the Fukushima Dai-Ichi Nuclear Power Plant Accident (TAC NO. MF0138), April 14, 2014 (ML14058B156).

[34] Limerick Generating Station, Unit 2- Staff Assessment of the Seismic Walkdown Report Supporting Implementation of Near-Term Task Force Recommendation 2.3, Related to the Fukushima Dai-Ichi Nuclear Power Plant Accident (TAC NO. MF0139), April 14, 2014 (ML14058B120).

[35] Limerick Generating Station, Units 1 and 2, Mitigating Strategies Assessment (MSA) Report for the Reevaluated Seismic Hazard Information - NEI 12-06, Appendix H, Revision 2, H.4.2 Path 2: GMRS < 2x SSE with High Frequency Exceedances, December 1, 2016 (ML16336A442).

[36] Limerick Generating Station, Units 1 And 2 - Staff Review of Mitigation Strategies Assessment Report of the Impact of the Reevaluated Seismic Hazard Developed In Response to the March 12, 2012, 50.54(f) Letter , April 12, 2017 (ML17087A066).

[37] Limerick Generating Station, Units 1 and 2, High Frequency Supplement to Seismic Hazard Screening Report, Response to NRC Request for Information Pursuant to 10 CFR 50.54(f), "Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," November 28, 2016 (ML16333A084).

[38] Limerick Generating Station, Units 1 and 2 - Staff Review of High Frequency Confirmation Associated With Reevaluated Seismic Hazard Implementing Near-Term Task Force Recommendation 2.1 , February 6, 2017 (ML17031A415).

[39] Calvert Cliffs Nuclear Power Plant, Units 1 and 2, Renewed Facility Operating License Nos. DPR-53 adn DPR-69, Docket Nos. 50-317 and 50-318, "Response to Request for Additional Information Regarding the Application to Adopt 10 CFR 50.69, 'Risk-informed categorization and treatment of structures, systems, and components for nuclear power reactors'," dated July 1, 2019 (ML19183A012).

[40] Calvert Cliffs Nuclear Power Plant, Units 1 and 2, Renewed Facility Operating License Nos. DPR-53 adn DPR-69, Docket Nos. 50-317 and 50-318, "Response to Request for Additional Information Regarding the Application to Adopt 10 CFR 50.69, 'Risk-informed categorization and treatment of structures, systems, and components for nuclear power reactors'," July 19, 2019 (ML19200A216).

[41] Calvert Cliffs Nuclear Power Plant, Units 1 and 2, Renewed Facility Operating License Nos. DPR-53 adn DPR-69, Docket Nos. 50-317 and 50-318, "Revised Response to Request for Additional Information Regarding the Application to Adopt 10 CFR 50.69, 'Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors,'letter dated July 19, 2019," dated August 5, 2019 (ML19217A143).

[42] Limerick Generating Station, Units 1 and 2, Response to Request for Additional Information: Application to Adopt 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components (SSCs) for Nuclear Power Plants," January 19, 2018 (ML18019A091).

[43] Limerick Generating Station, Units 1 and 2, Renewed Facility Operating License Nos. NPF-39 and NPF-85, NRC Docket Nos. 50-352 and 50-353, "Response to Request for Additional Information Application to Adopt 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components (SSCs) for Nuclear Power Plants," April 23, 2018 (ML18113A870).

[44] 032156-RPT-001, Limerick Generating Station PRA Finding Level Fact and Observation Technical Review, Revision 0, August 2016.

[45] 032362-RPT-10, Limerick Generating Station PRA Focused-Scope Peer Review & Finding Level Fact and Observation Independent Assessment, Revision 0, October 2018.

Appendix A
LGS FPIE PRA (Including Internal Flood) Findings and Disposition as Related to the 50.69 Alternate Defense-in-Depth and Pressure Boundary Processes

Finding ID (1)	Originating SR (2)	Finding	Status	Disposition	Impact to 50.69 Alternate Categorization Application
QU-F5-01	QU-F5 (Not Met)	Provide a discussion for the limitations of the quantification process that could impact applications (e.g., online maintenance, MPSI). One of the topics could be the WinNUPRA code limitations on the maximum number of cutsets and its impact on quantification truncation limits. (See also F&O for SY-B2.)	ADDRESSED	The LGS FPIE PRA summary and quantification notebook discusses the quantification process limitations on applications. This did not constitute an upgrade.	The documentation issue has been addressed. Therefore, there is no impact on the 50.69 alternate DID process or the EPRI pressure boundary methodology.
QU-F6-01	QU-F6 (Not Met)	Other than for HRA, the LGS documentation does not include the applied definition of "significant". Based on the review, the definitions provided in the ASME PRA Standard appear to have been generally applied.	ADDRESSED	The definitions in the PRA Standard have been added to the PRA model documentation. This did not constitute an upgrade.	The documentation issue has been addressed. Therefore, there is no impact on the 50.69 alternate DID process or the EPRI pressure boundary methodology .
IF-B3-01	IF-B3 Now IFSO-A5 (Not Met)	Basis for Significance: Since flood areas are documented as screened based on limited system volume, additional scenarios may need to be considered in the PRA if the system volume is considered. Discussion of Issue: Analysis of the TECW, RECW, CECW, and DWCW only considers the volume of water in the surge tank, not total system volume. Any system breach would result in gravity draining the system until level reaches that of the break. The TECW and RECW could contain significant volumes such that the scenarios may not be screened. Similarly, a break in the chilled water systems could release more water than in the surge tank. The DECW and RECW systems have automatic makeup to the surge tanks which could add water to the flood source.	ADDRESSED	The documentation and flood scenarios were reviewed and updated to address the open issues. Flood scenarios were screened based on hazard which includes a combination of flood source volume and if equipment in the area can be failed by the flood. This did not constitute an upgrade.	This finding has been addressed. The internal flood documentation and flood scenarios were reviewed and updated to address the issues. Therefore, there is no impact on the EPRI pressure boundary methodology Additionally, internal flood scenarios are not considered as part of the 50.69 alternate DID process.

Appendix A

LGS FPIE PRA (Including Internal Flood) Findings and Disposition as Related to the 50.69 Alternate Defense-in-Depth and Pressure Boundary Processes

Finding ID (1)	Originating SR (2)	Finding	Status	Disposition	Impact to 50.69 Alternate Categorization Application
IF-C2a-01	IF-C2a Now IFSN-A3 (Met CC I/II/III)	Basis for Significance: It appears from a review of appendices E and F that major actions needed have been identified. Discussion of Issue: No automatic actions were identified as being credited for flood termination or mitigation. Operator actions that are credited with terminating or mitigating a flooding event are generally described in Appendix E. However, the specific actions, such as, "close valve, V-XX," are not described in detail. The analyses shown in Appendix E Reference the HRA performed in Appendix F.	ADDRESSED	Appendix F of the internal flood notebook documents the operator actions credited for internal flood initiators. Appendix F provides detailed plant response, cues, location, timing, and execution information for each credited action. Appendix F references the HRA notebook which provides the HEP calculation worksheets and further details regarding HFEs. This did not constitute an upgrade.	This finding has been addressed. This was a documentation issue and therefore there is no impact on the EPRI pressure boundary methodology. Additionally, internal flood scenarios are not considered as part of the 50.69 alternate DID process.
IF-C2b-01	IF-C2b Now IFSN-A4 (Not Met)	Basis for Significance: No specific analysis of drains appears to have been performed. Discussion of Issue: Appendix E appears to take credit for drains; however, calculation of drain capacity was not evident.	ADDRESSED	The current internal flooding analysis includes a specific analysis of the drain capacity of RB-FL09, the only area where drains are credited. Section E.5 of the internal flood notebook provides a discussion of flood scenarios in Flood Zone RB-FL09. A drain capacity of 60,000 gallons was estimated and credited based on discussion with engineers and review of plant drawings. A probabilistic estimate of drainage failure is provided to address uncertainties in the drainage capacity. This did not constitute an upgrade.	This finding has been addressed. The current internal flooding analysis includes a specific analysis of the drain capacity of RB-FL09, the only area where drains are credited. Therefore, there is no impact on the EPRI pressure boundary methodology. Additionally, internal flood scenarios are not considered as part of the 50.69 alternate DID process.

Appendix A

LGS FPIE PRA (Including Internal Flood) Findings and Disposition as Related to the 50.69 Alternate Defense-in-Depth and Pressure Boundary Processes

Finding ID (1)	Originating SR (2)	Finding	Status	Disposition	Impact to 50.69 Alternate Categorization Application
IF-C3b-01, IF-C3b-03	IF-C3b Now IFSN-A8 (Met CC I)	<u>IF-C3b-01</u> Basis for Significance: Evaluation of barrier unavailability could result in significantly different flood scenarios. Evaluation of barrier unavailability is required by RG 1.200. Discussion of Issue: No consideration of barrier unavailability due to maintenance and how such unavailability could affect flood scenarios was documented. <u>IF-C3b-03</u> Basis for Significance: IF-C3b requires to "IDENTIFY inter-area propagation through the normal flow path from one area to another via drain lines; and areas connected via back flow areas connected via back flow through drain lines involving through drain lines involving failed check valves, pipe and failed check valves, pipe and cable penetrations...etc.". Discussion of Issue: LG-PRA-012, section 3.3.2.1, page 3-10, first paragraph describes how the EDG rooms are independent by discussing on doors and the corridor. Drains and electrical penetrations that may exist between the EDG rooms. Also, drains between the CE, TE, and RE are not discussed.	ADDRESSED	Section 3.4.10 of internal flooding notebook documents impacts of barrier unavailability. Section 3.4.12 documents considerations of backflow in drains where credited. Section 3.4.13 documents considerations of inter-area propagation flow paths. Section 3.4.14 documents considerations of structural analysis of doors where credited. Section 2.2.11 documents considerations of backflow through drains. The analysis does not explicitly address water entering flood zones via backflow through the drain piping since there are check valves installed in the drains that service the ECCS rooms in the basement of the Reactor Enclosure that prevent propagation of water from one room to another. Also, most internal drain lines within the plant drain to the Radioactive Waste system, which was observed to have a storage capacity of over 60,000 gallons Thus, backflow through drain lines was not explicitly modeled. However, specific analysis for drain backflow or determination of the reliability of drain line check valves has been performed. This did not constitute an upgrade.	This finding has been addressed. The reliability of the drain check valves and backflow have been evaluated and documented in the PRA internal flooding notebook. Therefore, there is no impact on the EPRI pressure boundary methodology. Additionally, internal flood scenarios are not considered as part of the 50.69 alternate DID process.

Appendix A

LGS FPIE PRA (Including Internal Flood) Findings and Disposition as Related to the 50.69 Alternate Defense-in-Depth and Pressure Boundary Processes

Finding ID (1)	Originating SR (2)	Finding	Status	Disposition	Impact to 50.69 Alternate Categorization Application
IF-D1-01	IF-D1 Now IFEV-A1 (Not Met)	Basis for Significance: It appears that some internal flooding scenarios may have been associated with an inappropriate initiating event. Discussion of Issue: All flooding initiators are classified as either turbine trip or manual shutdown events as documented in Appendix D. The LGS model includes loss of service water. TECW, RECW, and AC switchgear as special initiating events. As shown in Appendix C, several service water breaks are included in the internal flooding analysis, yet it is not clear why the events, were developed as turbine trip events as opposed to loss of service water events. As discussed under SR IF-B3, flooding events involving TECW and RECW were screened based on limited system volume. When flooding involving TECW and RECW are reevaluated, this SR must be considered. The documentation does not describe why flooding events that cause a loss of switchgear are not evaluated as a loss of AC switchgear.	ADDRESSED	Consistent with the SR IFEV-A1, an evaluation of the flood sources and subsequent scenarios was performed to group the initiating events. The events are generally classified as initiators that include either a turbine trip or manual shutdown event, as appropriate, with the impact of the initiator implied to fail those SSCs that are influenced by both internal flooding and spray effects. Where necessary, sub-scenario frequencies were identified for specific components that were susceptible to nearby spray sources. That is, certain SSCs were considered vulnerable to only those nearby sources of water that could render that particular component unavailable, i.e., approximately 10 feet within a given spray source. The internal flood notebook documents the specific mapping of flood scenarios to support system initiating events where appropriate. This did not constitute an upgrade.	This finding has been addressed. Mapping to support system initiators where appropriate was performed and was documented.in the internal flooding notebook. Therefore, there is no impact on the EPRI pressure boundary methodology. Additionally, internal flood scenarios are not considered as part of the 50.69 alternate DID process.

Appendix A

LGS FPIE PRA (Including Internal Flood) Findings and Disposition as Related to the 50.69 Alternate Defense-in-Depth and Pressure Boundary Processes

Finding ID (1)	Originating SR (2)	Finding	Status	Disposition	Impact to 50.69 Alternate Categorization Application
IF-E1-01	IF-E1 Now IFQU-A1 (Not Met)	Basis for Significance: Since flooding events appear to be improperly categorized and no documentation of a sequence review for applicability was found, this is assigned as a Finding. Discussion of Issue: All flooding initiators are classified as either turbine trip or manual shutdown events as documented in Appendix D. The LGS model includes loss of service water. TECW, RECW, and AC switchgear as special initiating events. As shown in Appendix C, several service water breaks are included in the internal flooding analysis, yet it is not clear why the events, were developed as turbine trip events as opposed to loss of service water events. Had flooding sequences been reviewed for applicability, the appropriate accident sequence could have been associated with the proper internal initiating events group. No documentation of a sequence review was performed.	ADDRESSED	Consistent with the SR IFEV-A1, an evaluation of the flood sources and subsequent scenarios was performed to group the initiating events. The events are generally classified as initiators that include either a turbine trip or manual shutdown event, as appropriate, with the impact of the initiator implied to fail those SSCs that are influenced by both internal flooding and spray effects. Where necessary, sub-scenario frequencies were identified for specific components that were susceptible to nearby spray sources. That is, certain SSCs were considered vulnerable to only those nearby sources of water that could render that particular component unavailable, i.e., approximately 10 feet within a given spray source. . Each scenario table in Appendix D of the internal flood notebook indicates what initiator is applied (e.g., IETT, IERECW, IETMS, etc.). This did not constitute an upgrade.	This finding has been addressed. Mapping to support system initiators where appropriate was performed and was documented.in the internal flooding notebook. Therefore, there is no impact on the EPRI pressure boundary methodology. Additionally, internal flood scenarios are not considered as part of the 50.69 alternate DID process.

Appendix A

LGS FPIE PRA (Including Internal Flood) Findings and Disposition as Related to the 50.69 Alternate Defense-in-Depth and Pressure Boundary Processes

Finding ID (1)	Originating SR (2)	Finding	Status	Disposition	Impact to 50.69 Alternate Categorization Application
IF-E5a-01	IF-E5a Now IFQU-A6 (Not Met)	Basis for Significance: An assessment of existing HFEs is required by the standard. Discussion of Issue: No systematic assessment of the existing operator actions that are included in flood sequences was performed.	ADDRESSED	Appendix F of the current internal flooding notebook documents the flooding impact on existing HFEs and the basis for the impact. This did not constitute an upgrade.	This finding has been addressed. Appendix F of the current internal flooding notebook documents the flooding impact on existing HFEs and the basis for the impact. This was a documentation issue and therefore there is no impact on the EPRI pressure boundary methodology. Additionally, internal flood scenarios are not considered as part of the 50.69 alternate DID process.
IF-E7-01	IF-E7 Now IFQU-A10 (Not Met)	Basis for Significance: No review of the LERF sequences for applicability was performed. Discussion of Issue: No review or quantification of flood-related LERF sequences is performed or documented.	ADDRESSED	Section 4.2, Figure 4.2, and Figure 4.4 of the internal flood notebook provide results of flood-related LERF. Flood scenarios or initiators that contribute to LERF are provided. Figure ES-2A and Figure ES-2B of the summary notebook provide flood-related contributions to total LERF. Section 6.0, Appendix G, Appendix H, and Appendix I of quantification notebook provides the LERF quantification results (including internal flood). Flood-related cutsets are provided. Sequence contributions to flood-related LERF were quantified including potential containment failure mode contributions (e.g., containment isolation, containment bypass, etc.) to flood-related LERF. This did not constitute an upgrade.	This finding has been addressed. CDF and LERF results by flooding initiator are included in the internal flooding notebook. The internal flooding sequences are included in the contributions to overall results and accident sequences in the quantification and summary notebooks as part of the integrated internal events model results. This was a documentation issue and therefore there is no impact on the EPRI pressure boundary methodology. Additionally, internal flood scenarios are not considered as part of the 50.69 alternate DID process.

Notes to Appendix A

1. Each of the finding IDs that begin the characters IF are from the internal flood peer review. The other findings are from the internal events peer review.

2. The SR listed first is the applicable SR from the standard version the peer review was performed against RA-Sb-2005, and the second is the applicable SR from the current standard, RA-Sa-2009.

ENCLOSURE 2

License Amendment Request

Limerick Generating Station, Units 1 and 2
Docket Nos. 50-352 and 50-353

Application to Implement an Alternate Defense-in-Depth Categorization Process, an Alternate Pressure Boundary Categorization Process, and an Alternate Seismic Tier 1 Categorization Process in Accordance with the Requirements of 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors"

Proposed License Condition Mark-Ups

APPENDIX C

ADDITIONAL CONDITIONS
OPERATING LICENSE NO. NPF-39

Exelon Generation Company, LLC shall comply with the following conditions on the schedule noted below:

Amendment No.	Additional Conditions
~~230~~[XXX]	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 SSCs and their associated supports; and the results of non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards, i.e., seismic margin analysis (SMA) to evaluate seismic risk, and a screening of other external hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009; as specified in Unit 1 License Amendment No. 230 dated July 31, 2018.
	~~Exelon will complete the implementation items listed in Attachment 2 of Exelon letter to NRC dated April 23, 2018 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).

Replace with UNIT 1 FOL INSERT

UNIT 1 FOL INSERT

In addition, Exelon is approved to implement 10 CFR 50.69 using any of the following alternative processes for categorization of RISC-1, RISC-2, RISC-3, and RISC-4 SSCs: the defense-in-depth approach contained in PWROG-20015-NP; the passive pressure boundary categorization approach described in EPRI 3002015999; and the seismic approach as described in Exelon's submittal letter dated March 11, 2021, as specified in Unit 1 License Amendment No. [XXX] dated [DATE].

APPENDIX C

ADDITIONAL CONDITIONS
OPERATING LICENSE NO. NPF-85

Exelon Generation Company, LLC shall comply with the following conditions on the schedule noted below:

Amendment No.	Additional Conditions
~~193~~ [YYY]	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 (AN0-2) passive categorization method to assess passive component risk for Class 2 and Class 3 SSCs and their associated supports; and the results of non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards, i.e., seismic margin analysis (SMA) to evaluate seismic risk, and a screening of other external hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009; as specified in Unit 2 License Amendment No. 193 dated July 31, 2018.
	~~Exelon will complete the implementation items listed in Attachment 2 of Exelon letter to NRC dated April 23, 2018 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).

Replace with UNIT 2 FOL INSERT

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In addition, Exelon is approved to implement 10 CFR 50.69 using any of the following alternative processes for categorization of RISC-1, RISC-2, RISC-3, and RISC-4 SSCs: the defense-in-depth approach contained in PWROG-20015-NP; the passive pressure boundary categorization approach described in EPRI 3002015999; and the seismic approach as described in Exelon's submittal letter dated March 11, 2021, as specified in Unit 2 License Amendment No. [YYY] dated [DATE].