UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

July 31, 2018

Mr. Bryan C. Hanson
Senior Vice President
Exelon Generation Company, LLC
President and Chief Nuclear Officer
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: 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)

Dear Mr. Hanson:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment Nos. 230 and 193 to Renewed Facility Operating License Nos. NPF-39 and NPF-85 for the Limerick Generating Station, Units 1 and 2, respectively, in response to your application dated June 28, 2017, as supplemented by letters dated August 14, 2017; January 19, 2018; April 23, 2018; and July 27, 2018.

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 nuclear power reactors in accordance with Title 10 of the *Code of Federal Regulations* Section 50.69.

A copy of our related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

V. Sreenivas, Project Manager
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-352 and 50-353

Enclosures:
1. Amendment No. 230 to Renewed NPF-39
2. Amendment No. 193 to Renewed NPF-85
3. Safety Evaluation

cc: Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-352

LIMERICK GENERATING STATION, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 230
Renewed License No. NPF-39

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Exelon Generation Company, LLC (Exelon Generation Company), dated June 28, 2017, as supplemented by letters dated August 14, 2017; January 19, 2018; April 23, 2018; and July 27, 2018, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Renewed Facility Operating License as indicated in the attachment to this license amendment.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

James Danna

James G. Danna, Chief
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility Operating License

Date of Issuance: July 31, 2018

ATTACHMENT TO LICENSE AMENDMENT NO. 230

LIMERICK GENERATING STATION, UNIT 1

RENEWED FACILITY OPERATING LICENSE NO. NPF-39

DOCKET NO. 50-352

Replace the following pages of the Renewed Facility Operating License with the attached revised pages. The revised pages are identified by amendment number and contain a marginal line indicating the area of change.

Remove	Insert
Page 5	Page 5
Appendix C, Page 1	Appendix C, Page 1

(16) Additional Conditions

The Additional Conditions contained in Appendix C, as revised through Amendment No. 230, are hereby incorporated into this renewed license. Exelon Generation Company shall operate the facility in accordance with the Additional Conditions.

(17) Exelon Generation Company shall provide to the Director of the Office of Nuclear Reactor Regulation a copy of any application, at the time it is filed, to transfer (excluding grants of security interests or liens) from Exelon Generation Company to its direct or indirect parent, or to any other affiliated company, facilities for the production, transmission, or distribution of electric energy having a depreciated book value exceeding ten percent (10%) of Exelon Generation Company's consolidated net utility plant, as recorded on Exelon Generation Company's books of account.

(18) Exelon Generation Company shall have decommissioning trust funds for Limerick, Unit 1, in the following minimum amount, when Limerick, Unit 1, is transferred to Exelon Generation Company:

Limerick, Unit 1	$94,127,446

(19) The decommissioning trust agreement for Limerick, Unit 1, at the time the transfer of the unit to Exelon Generation Company is effected and thereafter, is subject to the following:

(a) The decommissioning trust agreement must be in a form acceptable to the NRC.

(b) With respect to the decommissioning trust fund, investments in the securities or other obligations of Exelon Corporation or affiliates thereof, or their successors or assigns are prohibited. Except for investments tied to market indexes or other non-nuclear sector mutual funds, investments in any entity owning one or more nuclear power plants are prohibited.

(c) The decommissioning trust agreement for Limerick, Unit 1, must provide that no disbursements or payments from the trust shall be made by the trustee unless the trustee has first given the Director of the Office of Nuclear Reactor Regulation 30 days prior written notice of payment. The decommissioning trust agreement shall further contain a provision that no disbursements or payments from the trust shall be made if the trustee receives prior written notice of objection from the NRC.

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	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).

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-353

LIMERICK GENERATING STATION, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 193
Renewed License No. NPF-85

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Exelon Generation Company, LLC (Exelon Generation Company), dated June 28, 2017, as supplemented by letters dated August 14, 2017; January 19, 2018; April 23, 2018; and July 27, 2018, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Renewed Facility Operating License indicated in the attachment to this license amendment.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

James G. Danna, Chief
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility Operating License

Date of Issuance: July 31, 2018

ATTACHMENT TO LICENSE AMENDMENT NO. 193

LIMERICK GENERATING STATION, UNIT 2

RENEWED FACILITY OPERATING LICENSE NO. NPF-85

DOCKET NO. 50-353

Replace the following pages of the Renewed Facility Operating License with the attached revised pages. The revised pages are identified by amendment number and contain a marginal line indicating the area of change.

Remove	Insert
Page 8	Page 8
Page 9	Page 9
---	Appendix C, Page 1

(13) The licensee's UFSAR supplement submitted pursuant to 10 CFR 54.21(d), as revised during the license renewal application review process, and as revised in accordance with license condition 2.C.(12), describes certain programs to be implemented and activities to be completed prior to the period of extended operation (PEO).

(a) Exelon Generation Company shall implement those new programs and enhancements to existing programs no later than December 22, 2028.

(b) Exelon Generation Company shall complete those activities designated for completion prior to the PEO, as noted in Commitment Nos. 18, 19, 20, 22, 23, 24, 28, 29, 30, 38, 39, 40, 41, 42, 43, and 47, of Appendix A of NUREG-2171, "Safety Evaluation Report Related to the License Renewal of Limerick Generating Station, Units 1 and 2," no later than December 22, 2028, or the end of the last refueling outage prior to the period of extended operation, whichever occurs later.

(c) Exelon Generation Company shall notify the NRC in writing within 30 days after having accomplished item (a) above and include the status of those activities that have been or remain to be completed in item (b) above.

(14) The Additional Conditions contained in Appendix C, as revised through Amendment No. 193, are hereby incorporated into this renewed license. Exelon Generation Company shall operate the facility in accordance with the Additional Conditions.

D. The facility requires exemptions from certain requirements of 10 CFR Part 50 and 10 CFR Part 70. These include (a) exemption from the requirement of Appendix J, the testing of containment air locks at times when the containment integrity is not required (Section 6.2.6.1 of the SER and SSER-3), (b) exemption from the requirements of Appendix J, the leak rate testing of the Main Steam Isolation Valves (MSIVs) at the peak calculated containment pressure, Pa, and exemption from the requirements of Appendix J that the measured MSIV leak rates be included in the summation for the local leak rate test (Section 6.2.6.1 of SSER-3), (c) exemption from the requirement of Appendix J, the local leak rate testing of the Traversing Incore Probe Shear Valves (Section 6.2.6.1 of the SER and SSER-3), and (d) an exemption from the schedule requirements of 10 CFR 50.33(k)(l) related to availability of funds for decommissioning the facility (Section 22.1, SSER 8). The special circumstances regarding exemptions (a), (b) and (c) are identified in Sections 6.2.6.1 of the SER and SSER 3. An exemption from the criticality monitoring requirements of 10 CFR 70.24 was previously granted with NRC materials license No. SNM-1977 issued November 22, 1988. The licensee is hereby exempted from the requirements of 10 CFR 70.24 insofar as this requirement applies to the handling and storage of fuel assemblies held under this renewed license.

E. Deleted

F. The licensee shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

G. This renewed license is effective as of the date of issuance and shall expire at midnight on June 22, 2049.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

William M. Dean, Director
Office of Nuclear Reactor Regulation

Enclosures:

1. Appendix A - Technical Specifications
2. Appendix B - Environmental Protection Plan
3. Appendix C - Additional Conditions

Date of Issuance: October 20, 2014

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	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 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).

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 230 TO

RENEWED FACILITY OPERATING LICENSE NO. NPF-39 AND

AMENDMENT NO. 193 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-85

EXELON GENERATION COMPANY, LLC

LIMERICK GENERATING STATION, UNITS 1 AND 2

DOCKET NOS. 50-352 AND 50-353

1.0 INTRODUCTION

By letter dated June 28, 2017 (Reference 1), as supplemented by letters dated August 14, 2017 (Reference 2); January 19, 2018 (Reference 3); April 23, 2018 (Reference 4); and July 27, 2018 (Reference 5), Exelon Generation Company, LLC (Exelon, the licensee) submitted a license amendment request (LAR) for the Limerick Generating Station, Units 1 and 2 (Limerick). The licensee proposed to add a new license condition to the Renewed Facility Operating Licenses to allow the implementation of Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.69, "Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors." The provisions of 10 CFR 50.69 allow adjustment of the scope of structures, systems, and components (SSCs) subject to special treatment requirements (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation) based on a method of categorizing SSCs according to their safety significance.

By letter dated July 31, 2017 (Reference 6), and e-mails dated December 6, 2017 (Reference 7), and March 26, 2018 (Reference 8), the U.S. Nuclear Regulatory Commission (NRC, the Commission) staff ("the staff") requested additional information from the licensee. By letters dated August 14, 2017; January 19, 2018; April 23, 2018; and July 27, 2018, the licensee responded to the requests. The supplements provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on September 26, 2017 (82 FR 44854).

2.0 REGULATORY EVALUATION

2.1 Risk-Informed Categorization and Treatment of Structures, Systems, and Components

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

To take advantage of the safety enhancements available through the use of PRA, the NRC promulgated a new regulation, 10 CFR 50.69, in the *Federal Register* on November 22, 2004 (69 FR 68008), which became effective on December 22, 2004. The provisions of 10 CFR 50.69 allow adjustment of the scope of structures, systems, and components (SSCs) subject to special treatment requirements. Special treatment refers to those requirements that provide increased assurance beyond normal industry practices that SSCs perform their design-basis functions. For SSCs categorized as low safety-significance, alternative treatment requirements may be implemented in accordance with the regulation. For SSCs determined to be of high safety-significance, requirements may not be changed. This approach allows improved focus on equipment that has high safety-significance, resulting in improved plant safety.

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

Section 50.69 of 10 CFR does not allow for the elimination of SSC functional requirements or allow equipment that is required by the deterministic design basis to be removed from the facility. Instead, 10 CFR 50.69 enables licensees to focus their resources on SSCs that make a significant contribution to plant safety. For SSCs that are categorized as high safety-significant (HSS), existing treatment requirements are maintained or potentially enhanced. On the other hand, for SSCs categorized as low safety-significant (LSS) that do not significantly contribute to plant safety on an individual basis, the regulation allows an alternative risk-informed approach to treatment that provides a reasonable, although reduced, level of confidence that these SSCs will satisfy functional requirements. Implementation of 10 CFR 50.69 allows licensees to improve focus on equipment that has high safety-significance, resulting in improved plant safety.

2.2 Licensee's Proposed Changes

The licensee proposed to amend its Renewed Facility Operating Licenses by adding the following license condition that would allow for the implementation of 10 CFR 50.69:

> Exelon is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 structures, systems, and components (SSCs) using:
> Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events, including internal flooding, and internal fire; the shutdown safety

assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 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 the license amendment No. [XX] (Unit X) and No. [XX] (Unit X), dated [DATE], subject to the following condition:

. 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 PRA approach).

2.3 Regulatory Review

The NRC staff reviewed the licensee's application to determine whether (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) activities proposed 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 the health and safety of the public. The staff considered the following regulatory requirements and guidance during its review of the proposed changes.

Regulatory Requirements

Section 50.69 of 10 CFR provides an alternative approach for establishing requirements for treatment of SSCs for nuclear power reactors using a risk-informed method of categorizing SSCs according to their safety significance. Specifically, for SSCs categorized as low safety-significance, alternative treatment requirements may be implemented in accordance with the regulation. For SSCs determined to be of high safety-significance, requirements may not be changed.

Paragraph 50.69(c) of 10 CFR requires licensees to use an integrated decisionmaking process to categorize safety-related and non-safety-related SSCs according to the safety-significance of the functions they perform into one of the following four RISC categories, which are defined in 10 CFR 50.69(a), as follows:

RISC-1: Safety-related SSCs that perform safety-significant functions[1]

RISC-2: Non-safety-related SSCs that perform safety-significant functions

RISC-3: Safety-related SSCs that perform low safety-significant functions

RISC-4: Non-safety-related SSCs that perform low safety-significant functions

SSCs are classified as having either HSS functions (i.e., RISC-1 and RISC-2 categories) or low safety-significant (LSS) functions (i.e., RISC-3 and RISC-4 categories). For HSS SSCs, 10 CFR 50.69 maintains current regulatory requirements (i.e., it does not remove any requirements from these SSCs) for special treatment. For LSS SSCs, licensees can implement alternative treatment requirements in accordance with 10 CFR 50.69(b)(1) and 10 CFR 50.69(d). For RISC-3 SSCs, licensees can replace special treatment with an alternative treatment. For RISC-4 SSCs, 10 CFR 50.69 does not impose new treatment requirements, and RISC-4 SSCs are removed from the scope of any applicable special treatment requirements identified in 10 CFR 50.69(b)(1).

Paragraph 50.69(c)(1) of 10 CFR states that SSCs must be categorized as RISC–1, RISC–2, RISC–3, or RISC–4 SSCs using a categorization process that determines if an SSC performs one or more safety-significant functions and identifies those functions. The process must:

(i) Consider results and insights from the plant-specific PRA. This PRA must at a minimum model severe accident scenarios resulting from internal initiating events occurring at full power operation. The PRA must be of sufficient quality and level of detail to support the categorization process, and must be subjected to a peer review process assessed against a standard or set of acceptance criteria that is endorsed by the NRC.

(ii) Determine SSC functional importance using an integrated, systematic process for addressing initiating events (internal and external), SSCs, and plant operating modes, including those not modeled in the plant-specific PRA. The functions to be identified and considered include design bases functions and functions credited for mitigation and prevention of severe accidents. All aspects of the integrated, systematic process used to characterize SSC importance must reasonably reflect the current plant configuration and operating practices, and applicable plant and industry operational experience.

(iii) Maintain defense-in-depth.

(iv) Include evaluations that provide reasonable confidence that for SSCs categorized as RISC–3, sufficient safety margins are maintained and that any potential increases in core damage frequency (CDF) and large early

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[1] NEI 00-04 uses the term "high-safety-significant (HSS)" to refer to SSCs that perform safety-significant functions. The NRC understands HSS to have the same meaning as "safety-significant" (i.e., SSCs that are categorized as RISC-1 or RISC-2), as used in 10 CFR 50.69.

release frequency (LERF) resulting from changes in treatment permitted by implementation of § § 50.69(b)(1) and (d)(2) are small.

(v) Be performed for entire systems and structures, not for selected components within a system or structure.

Paragraph 50.69(c)(2) of 10 CFR states: "The SSCs must be categorized by an Integrated Decision-Making Panel (IDP) staffed with expert, plant-knowledgeable members whose expertise includes, at a minimum, PRA, safety analysis, plant operation, design engineering, and system engineering."

Paragraph 50.69(b)(3) of 10 CFR states that the Commission will approve a licensee's implementation of this section by issuance of a license amendment if the Commission determines that the categorization process satisfies the requirements of 10 CFR 50.69(c). As stated in 10 CFR 50.69(b), after the NRC approves an application for a license amendment, a licensee may voluntarily comply with 10 CFR 50.69 as an alternative to compliance with the following requirements for LSS SSCs: (i) 10 CFR Part 21, (ii) a portion of 10 CFR 50.46a(b), (iii) 10 CFR 50.49, (iv) 10 CFR 50.55(e), (v) certain requirements of 10 CFR 50.55a, (vi) 10 CFR 50.65, except for paragraph (a)(4), (vii) 10 CFR 50.72, (viii) 10 CFR 50.73, (ix) Appendix B to 10 CFR Part 50, (x) certain containment leakage testing requirements, and (xi) certain requirements of Appendix A to 10 CFR Part 100.

Guidance

Nuclear Energy Institute (NEI) 00-04, Revision 0, "10 CFR 50.69 SSC Categorization Guideline" (Reference 9), describes a process for determining the safety-significance of SSCs and categorizing them into the four RISC categories defined in 10 CFR 50.69. This categorization process is an integrated decisionmaking process that incorporates risk and traditional engineering insights. NEI 00-04, Revision 0, provides options for licensees implementing different approaches depending on the scope of their PRA models. It also allows the use of non-PRA approaches when PRAs have not been performed. NEI 00-04 identifies non-PRA approaches such as fire-induced vulnerability evaluation to address fire risk, seismic margin analysis (SMA) to address seismic risk, and guidance in Nuclear Management and Resource Council (NUMARC) 91-06, "Guidelines for Industry Actions to Assess Shutdown Management" (Reference 10), to address shutdown operations. As stated in Regulatory Guide (RG) 1.201, such non-PRA-type evaluations will result in more conservative categorization, in that special treatment requirements will not be allowed to be relaxed for SSCs that are relied upon in such evaluations. The degree of relief that the NRC will accept under 10 CFR 50.69 (i.e., SSCs subject to relaxation of special treatment requirements) will be commensurate with the assurance provided by the evaluation.

Sections 2 through 10 of NEI 00-04 describe a method for meeting the requirements of 10 CFR 50.69(c), as follows:

- Sections 3.2 and 5.1 provide specific guidance corresponding to 10 CFR 50.69(c)(1)(i).
- Sections 3, 4, 5 and 7 provide specific guidance corresponding to 10 CFR 50.69(c)(1)(ii).
- Section 6 provides specific guidance corresponding to 10 CFR 50.69(c)(1)(iii).
- Section 8 provides specific guidance corresponding to 10 CFR 50.69(c)(1)(iv).
- Section 2 provides specific guidance corresponding to 10 CFR 50.69(c)(1)(v).
- Sections 9 and 10 provide specific guidance corresponding to 10 CFR 50.69(c)(2).

Additionally, Section 11 of NEI 00-04 provides guidance on program documentation and change control related to the requirements of 10 CFR 50.69(e), and Section 12 of NEI 00-04 provides guidance on periodic review related to the requirements in 10 CFR 50.69(f). Maintaining change control and periodic review provides confidence that all aspects of the program reasonably reflect the current plant configuration and operating practices, and applicable plant and industry operational experience, as required by 10 CFR 50.69(c)(1)(ii).

RG 1.201 (For Trial Use), Revision 1, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants according to Their Safety Significance" (Reference 11), endorses the categorization method described in NEI 00-04, Revision 0, with clarifications, limitations, and conditions. RG 1.201 states that the applicant is expected to document, at a minimum, the technical adequacy of the internal initiating events PRA. Licensees may use either PRAs or alternative approaches for hazards other than internal initiating events. RG 1.201 clarifies that the NRC staff expects that licensees proposing to use non-PRA approaches in their categorization should provide a basis in the submittal for why the approach and the accompanying method employed to assign safety-significance to SSCs is technically adequate. It further states that as part of the NRC's review and approval of a licensee's or applicant's application requesting to implement 10 CFR 50.69, the NRC staff intends to impose a license condition that will explicitly address the scope of the PRA and non-PRA methods used in the licensee's categorization approach. If a licensee or applicant wishes to change its categorization approach and the change is outside the bounds of the NRC's license condition (e.g., switch from a seismic margins analysis to a seismic PRA), the licensee or applicant will need to seek NRC approval, via a license amendment, of the implementation of the new approach in their categorization process. RG 1.201 also states that all aspects of NEI 00-04 must be followed to achieve reasonable confidence in the evaluations required by 10 CFR 50.69(c)(1)(iv).

RG 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities" (Reference 12) describes an acceptable approach for determining whether the quality of the PRA, in total or the parts that are used to support an application, is sufficient to provide confidence in the results, such that the PRA can be used in regulatory decision making for light-water reactors. It endorses, with clarifications, the American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) PRA Standard ASME/ANS RA-Sa-2009 ("ASME/ANS 2009 Standard" or "PRA Standard") (Reference 13). This RG provides guidance for determining the technical adequacy of a PRA by comparing the PRA to the relevant parts of the ASME/ANS RA-Sa-2009 using a peer review process. In accordance with the guidance, peer reviews should be used for PRA upgrades. A PRA upgrade is defined in the PRA Standard as "the incorporation into a PRA model of a new methodology or significant changes in scope or capability that impact the significant accident sequences or the significant accident progression sequences."

RG 1.174, Revision 3, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" (Reference 14), provides guidance on the use of PRA findings and risk insights in support of changes to a plant's licensing basis. This RG provides risk acceptance guidelines for evaluating the results of such evaluations.

3.0 TECHNICAL EVALUATION

3.1 Staff's Method of Review

In determining whether an amendment to a license will be issued, the Commission is guided by the considerations that govern the issuance of initial licenses to the extent applicable and appropriate. The staff evaluated the licensee's application to determine if the proposed changes are consistent with the regulations and guidance discussed in Section 2 of this safety evaluation (SE). Paragraph 50.69(b)(3) of 10 CFR states that the Commission will approve a licensee's implementation of 10 CFR 50.69 by issuing a license amendment if it determines that the licensee's process for categorizing SSCs satisfies the requirements to 10 CFR 50.69(c). The staff reviewed the licensee's SSC categorization process against the categorization process described in NEI 00-04, Revision 0, as endorsed by RG 1.201, and against the requirements in 10 CFR 50.69(c). The staff's review and the documentation of that review in this SE uses the framework of NEI 00-04, Revision 0.

3.2 Overview of the Categorization Process (NEI 00-04, Section 2)

Sections 1.5 and 2 of NEI 00-04 provide an overview of the categorization process. RG 1.201 provides that the categorization process described in NEI 00-04, with any noted exceptions or clarifications, is acceptable for implementation of 10 CFR 50.69.

The licensee stated in the LAR that it will implement the risk categorization process in accordance with NEI 00-04, as endorsed by RG 1.201; however, the licensee provided little detail of the categorization process. Therefore, in Request for Additional Information (RAI) 05 (Reference 7), the staff requested the licensee to (1) summarize the categorization process, (2) provide the order of the sequence of elements or steps that will be performed, (3) explain the difference between preliminary HSS and assigned HSS, and (4) identify which inputs can and which cannot be changed by the IDP from preliminary HSS to LSS.

Table 1 provided in response to RAI 05 (Reference 3), as further revised in response to followup RAI 05.01 (Reference 4), summarizes the process and is shown below.

In response to RAI 05, the licensee summarized the categorization process and described which steps are performed at the component level and which steps are performed at the function level. The licensee explained that the execution sequence of steps/elements of the process does not impact the resulting preliminary categorization because the safety determination of each element of the process is independent of each other.

As summarized in the licensee's response to RAI 05, the categorization process contains the following elements/steps:

- Defining system boundaries (see Section 3.3 of this SE).
- Defining system function and assigning components to functions (see Section 3.4 of this SE).
- Risk Characterization. Safety-significance of active components is assessed through a combination of PRA and non-PRA methods, covering all hazards (see Section 3.5 of this SE).

- Defense-in-depth (DID) characterization performed in accordance with Section 6 of NEI 00-04 (see Section 3.6 of this SE).

- Passive Characterization. Passive components are not modeled in the PRA, and therefore, a different assessment method is used to assess the safety-significance of these components, as described in Section 3.5.4 of this SE. This process addresses those components that have only a pressure-retaining function and the passive function of active components, such as the pressure/liquid retention of the body of a motor-operated valve.

- Qualitative Characterization. System functions are qualitatively categorized as HSS or LSS based on the seven questions in Section 9.2 of NEI 00-04 (see Section 3.9 of this SE).

- Cumulative risk sensitivity study. For PRA-modeled components, an overall risk sensitivity study is used to confirm that the population of LSS components results in acceptably small increases to CDF and LERF and meets the acceptance guidelines of RG 1.174 (see Section 3.8 of this SE).

- Review by the IDP. The categorization results are presented to the IDP for review and approval. The IDP reviews the categorization results and makes the final determination on the safety-significance of system functions and components (see Section 3.9 of this SE).

In response to RAI 05.b, the licensee explained that consistent with NEI 00-04, the categorization of a component or function is "preliminary" until it has been confirmed by the IDP (see also Section 3.9 of this SE). The licensee stated that a component or function is preliminarily categorized as HSS if any element of the process results in a preliminary HSS determination. This preliminary categorization will be presented to the IDP for review. The IDP will decide the final categorization as further discussed in Section 3.9 of this SE.

In Table 1 of the RAI response, and in response to the NRC's staff request in RAI 05.c, the licensee provided clarifications on how some steps of the process are performed at the component level (e.g., all PRA and non-PRA-modeled hazards, containment DID, passive categorization), how some steps are performed at the function level (e.g., qualitative criteria), and how some steps are performed at the function and component level (e.g., shutdown, core damage DID).

As further discussed in Section 3.7 of this SE, if any SSC is identified as HSS from either the PRA component safety-significance assessment (internal events in Section 5.1 of NEI 00-04, integral PRA assessment in Section 5.6 of NEI 00-04) or the DID assessment (Section 6 of NEI 00-04), the associated system function(s) would be identified as HSS. Once a system function is identified as HSS, then all the components supporting that function are preliminary HSS and will be presented to the IDP for review.

The NRC staff has evaluated the categorization steps and the associated clarifications provided by the licensee in response to RAI 05 and Table 1 of the RAI 05 response, as further revised in response to followup RAI 05.01, and finds that the licensee's process is consistent with all aspects of the process in NEI 00-04, as endorsed by RG 1.201.

Table 1

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

3.3 Assembly of Plant-Specific Inputs (NEI 00-04, Section 3)

Paragraph 50.69(c)(1)(ii) of 10 CFR requires licensees to determine SSC functional importance using an integrated, systematic process for addressing initiating events (internal and external), SSCs, and plant operating modes, including those not modeled in the plant-specific PRA. The functions to be identified and considered include design-basis functions and functions credited for mitigation and prevention of severe accidents. Section 4 of NEI 00-04 provides guidance for developing a systematic engineering assessment involving the identification and development of base information necessary to perform the risk-informed categorization. The assessment

[2] As further discussed in Section 3.9 of this SE, the licensee explained in response to followup RAI 05.01 that the seven qualitative criteria are assessed preliminary by the 10 CFR 50.69 categorization team prior to the IDP. The licensee further clarified that if the IDP determines that any one of the seven qualitative criteria cannot be confirmed (false response) for a system function, then the final categorization of that function will be HSS.

includes the following elements: system selection and system boundary definition, identification of system functions, and a mapping of components to functions.

Section 4 of NEI 00-04 states that system selection and boundary definition include defining system boundaries where the system interfaces with other systems. NEI 00-04 states that the next step is the identification of system functions, including design basis and beyond design-basis functions identified in the PRA, and that system functions should be consistent with the functions defined in design-basis documentation and maintenance rule functions. NEI 00-04 states that the coarse mapping of components to functions involves the initial breakdown of system components into system functions they support. The licensee should then identify and document system components and equipment associated with each function.

Paragraph 50.69(c)(1)(v) of 10 CFR requires that categorization be performed for entire systems and structures, not for selected components within a system or structure. The process described in the LAR and summarized above is consistent with, and capable of, collecting and organizing information at the system level by defining boundaries, functions, and components. Therefore, the NRC staff finds that 10 CFR 50.69(c)(1)(v) will be satisfied upon implementation of the licensee's 10 CFR 50.69 categorization process.

3.4 System Engineering Assessment (NEI 00-04, Section 4)

Section 2.2 of the LAR states that the safety functions in the categorization process include the design-basis functions, as well as functions credited for severe accidents (including external events). Section 3.1.1 of the LAR summarizes the different hazards and plant states for which functional and risk-significant information will be collected. Section 3.1.1 of the LAR also states that the SSC categorization process documentation will include, among other items, system functions identified and categorized with the associated bases and mapping of components to support function(s).

Paragraph 50.69(c)(1)(ii) of 10 CFR requires, in part, that the functions to be identified and considered in the categorization process include design-basis functions and functions credited for mitigation and prevention of severe accidents. NEI 00-04 includes guidance to identify all functions performed by each system and states that the IDP will categorize all system functions. All system functions include all functions involved in the prevention and mitigation of accidents and may include additional functions not credited as hazard mitigating functions, depending on the system. The LAR summarizes the applicable guidance in NEI 00-04 and states that the guidance in NEI 00-04 will be followed. Therefore, the NRC staff finds that the licensee described a systematic process that will identify design-basis functions and functions credited for mitigation and prevention of severe accidents, consistent with the requirements of 10 CFR 50.59(c)(1)(ii), because all system functions will be identified and evaluated through the categorization process, in accordance with NEI 00-04.

3.5 Component Safety-Significance Assessment (NEI 00-04, Section 5)

This step in the licensee's categorization process is to assess the safety-significance of components using quantitative or qualitative risk information from a PRA or other risk assessment method. In the NEI 00-04 guidance, component risk significance is assessed separately for five hazard groups:

- Internal event risk
- Fire

- Seismic
- Other external risks (tornadoes, external floods)
- Shutdown risks

Paragraph 50.69(c)(1)(i) of 10 CFR requires, in part, the use of PRA to assess risk from internal events as a minimum. The paragraph further specifies that the PRA used in the categorization process must be of sufficient quality and level of detail and subject to an acceptable peer review process. For the hazards other than internal events, including fire, seismic, other external hazards (high winds, external floods, etc.), and shutdown, 10 CFR 50.69(b)(2) allows, and the NEI 00-04 guidance summarizes, the use of PRA if such PRA models exist, or, in the absence of quantifiable PRA, the use of other methods (e.g. fire-induced vulnerability evaluation, SMA, individual plant examination of external events (IPEEE) screening, and shutdown safety plan).

LAR Sections 3.1.1 and 3.2.1 through 3.2.5 explain that the licensee's categorization process uses PRA to assess risks from internal events (including internal flooding) and from fire. For the other three risk hazard groups, the licensee's process uses non-PRA methods for the risk characterization, as follows:

- SMA to assess seismic risk
- IPEEE screening to assess the risk from other external hazards (high winds, external floods)
- Shutdown safety plan to assess shutdown risk

The methods used by the licensee to assess internal and external hazards are consistent with the methods included in the NEI 00-04 guidance, as endorsed by RG 1.201, and therefore, acceptable to the NRC staff. The guidance considers the results and insights from the plant-specific PRA peer reviews as required by 10 CFR 50.69(c)(1)(i), and non-PRA risk characterization as required by 10 CFR 50.69(c)(1)(ii). The application of these methods is reviewed in the following SE subsections: PRA in Subsections 3.5.1 and 3.5.2, and the non-PRA methods in Subsection 3.5.3.

3.5.1 Capability and Quality of the PRA to Support the Categorization Process

The licensee's PRA is comprised of (1) an internal events PRA that calculates CDF and LERF from internal events, including internal flooding, at full power, and (2) a fire PRA. Paragraph 50.69(c)(1)(i) of 10 CFR requires, in part, that the PRA must be of sufficient quality and level of detail to support the categorization process, and must be subjected to a peer review process assessed against a standard or set of acceptance criteria that is endorsed by the NRC. Paragraph 50.69(b)(2)(iii) of 10 CFR requires the results of the PRA review process conducted to meet 10 CFR 50.59(c)(1)(i) be submitted as part of the application. The licensee has submitted this information and the NRC staff's review of this information is presented below.

Internal Events PRA

The NRC staff's review of the internal events and flooding PRAs was based on the results of the peer review of the internal events and flooding PRAs; the associated facts and observations (F&O) closure review described in LAR Sections 3.2.1 and 3.3 and presented in LAR Attachment 3; and previously docketed information on PRA quality submitted to the NRC in the relief request for risk-informed inservice inspection (ISI) dated April 13, 2016 (Reference 15), and September 19, 2016 (Reference 16). The last full-scope peer review of the internal events PRA was performed in 2005 against RG 1.200, Revision 0. The internal flooding PRA was peer reviewed in 2008 against RG 1.200, Revision 1 (Reference 17). Further, the licensee performed gap assessments of the internal events and internal flooding PRA against the PRA standard, ASME/ANS 2009 Standard, as endorsed by RG 1.200, Revision 2. The gap assessments of the internal events PRA were provided in the licensee's response to RAI 02 (Reference 16) related to the risk-informed ISI relief request, and the gap assessment for the internal flooding PRA was provided in the LAR supplement dated August 14, 2017. The gap assessment concluded that there were no deficiencies in the internal events or internal flooding PRAs that were not previously identified in an F&O.

In the LAR, the licensee stated that in July 2016, an F&Os closure review was performed by an independent team on all internal events and internal flooding finding-level F&Os. This July 2016 F&O closure review was a pilot review to develop the process to be detailed in Appendix X (Reference 18) to the guidance in NEI 05-04 (Reference 19), NEI 07-12 (Reference 20), and NEI 12-13 (Reference 21), concerning the process "Close-Out of Facts and Observations." The NRC staff accepted, with conditions, a final version of Appendix X to NEI 05-04, 07-12, and 12-13 in the NRC letter dated May 3, 2017 (Reference 22), which differed from the guidance used by the licensee in the July 2016 F&O closure. Consequently, in LAR Attachment 3, Tables 3a and 3b, the licensee submitted all the F&Os from the peer reviews, including the F&Os that were considered resolved by the F&O closure review. For each F&O, the licensee provided a disposition for this application.

The NRC staff reviewed the licensee's resolution of all the peer review findings and assessed the potential impact of the findings on the categorization. The NRC staff requested additional information to clarify the licensee's disposition for some of the findings, as described in the following paragraphs.

Open F&O HR-A1-01 found that the test and maintenance pre-initiators were not derived from a review of procedures and practices as stated in supporting requirement (SR) HR-A1 of the PRA standard. In RAI 01.a (Reference 7), the NRC staff requested that the licensee provide an explanation of how the test and maintenance pre-initiators were derived and provide justification that the risk-significant pre-initiators have been included in the PRA model. In response to RAI 01.a (Reference 3), the licensee stated that the PRA model includes numerous pre-initiators for a number of risk-significant systems, but these were not derived from a formal review of procedures and practices. In response to RAI 04 (Reference 3) and followup RAI 04.01 (Reference 4), the licensee proposed implementation item i to update the human reliability analysis (HRA) pre-initiators in the internal events PRA model to meet Capability Category II (CC-II) of the ASME/ANS 2009 Standard, as endorsed by RG 1.200, Revision 2; conduct a focused-scope peer review of the pre-initiator analysis; and resolve any resulting F&Os prior to implementation of the 10 CFR 50.69 categorization (see Section 3.5.5 of this SE). The NRC staff concludes this issue is resolved because, prior to implementation of the 10 CFR 50.69 categorization, the HRA pre-initiators in the internal events PRA model will undergo a focused-scope peer review and any associated finding-level F&Os will be resolved to meet CC-II. Furthermore, the scope of this peer review is narrow and the categorization process established under NEI 00-04, including the HRA sensitivity studies required by Table 5-2 of NEI 00-04, are expected to address the impact of the HRA pre-initiator uncertainties on the 10 CFR 50.69 categorization results.

Open F&O IF-B3-01 found that internal flooding scenarios may have been inappropriately screened out because the full volume of water that could drain from the Turbine Enclosure

Cooling Water, Reactor Enclosure Cooling Water, Control Enclosure Cooling Water, and Drywell Chilled Water systems was not considered. The disposition to this F&O states that flood scenarios need to be reviewed in the next PRA update to determine if revisions or additional scenarios are needed. The licensee also stated that any changes are expected to have "no material impact" on the 10 CFR 50.69 application. In RAI 01.b (Reference 7), the NRC staff requested the licensee to either justify why the excluded scenarios have no impact on the 10 CFR 50.69 application or to incorporate the additional water volumes cited in the F&O into the internal flooding PRA. In response to RAI 01.b (Reference 3), the licensee explained that contrary to the F&O statement, these systems were not screened out of the internal flooding PRA model and that the PRA includes all identified water sources and their scenario frequencies; therefore, the NRC staff finds that this has no impact on the application.

Resolved F&O SC-SY-B1-01 stated that "a high probability was used for failure of fire water makeup to the vessel to prevent core damage ... to include the uncertainty as to whether or not the fire protection system can actually prevent core damage after depressurizing the reactor and within four hours after an initiating event." The resolution to the F&O stated that a "detailed HRA [human reliability analysis] calculation was performed for aligning fire water makeup to the reactor vessel." Performing a "detailed HRA calculation" provides confidence that the alignment is feasible and that failure to align is properly quantified. This calculation does not necessarily evaluate whether the flow and amount of water is sufficient to prevent core damage. The NRC staff requested the licensee in RAI 01.c (Reference 7) to provide justification that the fire protection system can successfully prevent core damage as credited in the PRA model. In response to RAI 01.c (Reference 3), the licensee explained that the PRA documentation confirms the ability of the fire water system to provide sufficient water flow to the reactor vessel when the pressure of the reactor vessel is 100 pounds per square inch gauge or less, consistent with the assumptions for the HRA calculation. The licensee further explained that the fire water injection source is modeled as a success path in the PRA in late scenarios when the reactor pressure vessel has been depressurized. Because the licensee confirmed that the fire water system can provide sufficient flow and amount of water to prevent core damage, the NRC staff concludes that this F&O has no impact on the application.

Resolved F&Os DA-C14-01 and QU-A4-01 state that credit for repair had been removed for some listed systems but implied that credit was retained for the instrument air system. The NRC staff noted that this statement appears to contradict the statements in the internal events gap assessment 15 – that "repair is not credited in the current model." Therefore, in RAI 01.d (Reference 7), the NRC staff requested the licensee to identify any SSCs for which repair is credited in the current internal events PRA and justify how the applicable SRs from the PRA standard (e.g., SY-A24, DA-C1, LEC3, and CA-C15) are met. In response to RAI 01.d (Reference 3), the licensee clarified that the recovery of instrument air is credited in the PRA model and proposed implementation item ii to remove the credit for recovery of instrument air from the internal events PRA model prior to implementation of the 10 CFR 50.69 categorization process (see Section 3.5.5 of this SE).

There have been PRA updates since the 2005 peer review of the internal events PRA, including updates to resolve the F&Os. In its response to RAI 01 (Reference 16) associated with the risk-informed ISI relief request (Reference 15), the licensee provided identification of PRA updates that have been made since the most recent peer review in 2005 and provided a description of the changes. The licensee concluded that the changes constitute PRA maintenance and not PRA upgrades that would require a focused-scope peer review. The NRC staff found that one of the 2008 changes was to convert HRA from a spreadsheet to the Electric Power Research Institute (EPRI) HRA Calculator. This change was stated to be PRA maintenance because the "HRA Calculator uses the same HRA methodologies as were used in the spreadsheets."

In RAI 03.b (Reference 7), the staff requested that the licensee provide justification for why the change to the use of the HRA Calculator was not an upgrade that would require a focused-scope peer review. In response to RAI 03.b (Reference 3), the licensee summarized the methods used for post-initiator actions and dependency analysis prior to implementation of the EPRI HRA Calculator and clarified that the methods are the same as those used by implementation of the HRA Calculator. The licensee concluded that the transition to the EPRI HRA Calculator from a spreadsheet approach did not constitute a method change. The response also stated that methods used for pre-initiator actions are also the same but that pre-initiator actions have not been transitioned to the HRA Calculator. Additionally, the licensee provided a summary of the differences in human error probabilities between the 2004 and 2008 PRA models for 17 independent, risk-significant operator actions and justified that the numerical human error probabilities differences were mainly due to using the mean values from technique for human error rate prediction, as opposed to the median values that were used in the spreadsheet. Because the licensee confirmed that the collection of HRA methods used was the same as prior to implementing the EPRI HRA Calculator and showed that the HRA values for risk-significant post-initiator operator actions were similar, the NRC finds acceptable the licensee's evaluation that the implementation of the HRA Calculator is not a PRA upgrade, and therefore, a focused-scope peer review is not required.

RG 1.200 provides guidance for determining the technical adequacy of the PRA by comparing the PRA to the relevant parts of the ASME/ANS 2009 Standard using a peer review process. The licensee has followed the guidance and submitted the results of the peer review, and therefore, met 10 CFR 50.69(b)(2)(iii). The NRC staff has reviewed the peer review results and the licensee's resolution of the results and finds that the quality and level of detail of the PRA is sufficient to support the categorization of SSCs using the process endorsed by the NRC staff in RG 1.201. Significant errors and weaknesses in the internal events PRA will be resolved prior to implementation of the 10 CFR 50.69 categorization process with the completion of implementation items i, ii (discussed in this section of the SE), vii, viii, and ix (discussed in Section 3.5.2 of this SE). Therefore, the NRC staff concludes that the internal events PRA with the completion of the proposed implementation items i, ii, vii, viii, and ix, meets the internal events PRA requirement in 10 CFR 50.69(c)(1)(i).

Fire PRA

The NRC staff reviewed the results of the peer review of the fire PRA and associated F&O closure review described in LAR Sections 3.2.2 and 3.3 and presented in LAR Attachment 3. The licensee's fire PRA was subject to a full-scope industry peer review in November 2011 against RG 1.200, Revision 2.

In July 2016, an F&Os closure review was performed by an independent team on fire events finding-level F&Os. The July 2016 F&O closure review was a pilot review to develop the process to be detailed in Appendix X to the guidance in NEI 05-04, NEI 07-12, and NEI 12-13, concerning the process "Close Out of Facts and Observations." The staff accepted, with conditions, a final version of Appendix X in the letter dated May 3, 2017, which differed from the guidance used by the licensee in the July 2016 F&O closure. Consequently, in LAR Attachment 3, Tables 3a and 3b, the licensee submitted all the F&Os from the peer reviews, including the F&Os that were considered resolved by the F&O closure review. For each F&O, the licensee provided a disposition for this application.

The NRC staff reviewed the licensee's resolution of all the peer review findings and considered the potential impact of the findings on the 10 CFR 50.69 categorization. The NRC staff requested additional information to clarify the licensee's disposition for some of the findings, as described in the following paragraphs.

Disposition to open F&O 2-8 related to SR PRM-B6 stated, "The characterization of a MSIV [main steam isolation valve] spurious opening as a LLOCA [large loss-of-coolant accident] above TAF [top of active fuel] was not supported by T/H [thermal-hydraulic]." The licensee stated that this F&O is a "documentation issue with no impact on the application." Because this F&O appeared to identify potential modeling issues in the PRA model, the NRC staff requested the licensee in RAI 02.a (Reference 7) to provide justification that appropriate success criteria have been used in the PRA model for scenarios that consider MSIV spurious opening such as a large loss-of-coolant accident. In response to RAI 02.a (Reference 3) and followup RAI 04.01 (Reference 4), the licensee proposed implementation item iii to update the success criteria for MSIV spurious opening prior to implementation of the 10 CFR 50.69 categorization process (see Section 3.5.5 of this SE).

Open F&O 2-25 related to SR FSS-D7 found that there is no documentation to verify that the credited fire detection and suppression system is installed and maintained in accordance with applicable codes and standards, and that the credited system is in a fully operable state during plant operation. The F&O disposition only addressed the availability and/or reliability of the fire detection and suppression system(s). Therefore, in RAI 02.b (Reference 7), the NRC staff requested the licensee to either verify that all credited fire detection and suppression systems have been reviewed for compliance to applicable codes and documented in a code-compliance calculation, or otherwise, to evaluate all credited fire detection and suppression systems against the original code of construction, and adjust credit in the PRA accordingly. In response to RAI 02.b (Reference 3), the licensee stated that a review of the fire protection program was performed for the fire PRA credited fire detection and suppression systems. The licensee explained that this review included the Fire Protection System Design Baseline Document, the Updated Final Safety Analysis Report Section 9.5.1 (Fire Protection program), and Updated Final Safety Analysis Report Appendix 9A (Fire Protection Evaluation Report), which detail the design of the fire protection systems in accordance with the applicable codes. The licensee concluded from this review that the credited fire PRA fire detection and suppression systems were installed and maintained in accordance with the applicable codes and standards. Additionally, the licensee explained that the licensee's procedures and work management processes ensure that the fire PRA credited systems are maintained in accordance with the applicable codes. Therefore, the NRC staff concludes that this F&O has no impact on the application.

In disposition to open F&O 4-6 related to SR HRA-A3, the licensee stated that undesired operator actions, such as tripping or isolating equipment, were identified that could result from spurious signals, but such actions were determined to have "no material impact" on the 10 CFR 50.69 application. The disposition explains that in such cases, there would be time for recovery if there is no damage to equipment from fire. Because undesired operator actions based on spurious signals create additional risk, and the success of recovery actions can be hampered by fire or fire damage and the difficulty of diagnosing what is happening in the plant when spurious signals have occurred, in RAI 02.c (Reference 7), the NRC staff requested the licensee to provide justification for not modeling undesired operator actions in the fire PRA. In response to RAI 02.c (Reference 3) and followup RAI 04.01 (Reference 4), the licensee proposed implementation item iv to model undesired operator actions in the fire PRA, conduct a focused-scope peer review, and resolve any F&Os prior to implementation of the 10 CFR 50.69 categorization process. In response to followup RAI 02.01 (Reference 4), the licensee stated that NUREG-1921, "EPRI/NRC-RES Fire Human Reliability Analysis Guidelines" (Reference 23), methodology was applied in identifying undesired operator actions and will be used to incorporate the actions into the fire PRA (see Section 3.5.5 of this SE). The NRC staff concludes this issue is resolved because, prior to implementation of the 10 CFR 50.69 categorization, the modeling of undesired operator actions in the fire PRA will undergo a focused-scope peer review and any associated finding-level F&Os will be resolved. Furthermore, the scope of this peer review is narrow, the licensee will follow NRC-accepted guidance to model undesired operator actions in the fire PRA, and the categorization process established under NEI 00-04, including the HRA sensitivity studies required by Table 5-3 of NEI 00-04, are expected to address the impact of the HRA uncertainties on the 10 CFR 50.69 categorization results.

Closed F&O 4-23 relates to SR PRM-C1 and identifies that "MCRAB [Main Control Room Abandonment] Event tree uses existing FPIE [full power internal events] success criteria and T-H [thermal-hydraulic] analysis," without the proper confirmation or justification of applicability. The F&O resolution stated that, "In general, the use of internal events and/or fire non-abandonment T/H runs for MCRAB actions is appropriate when the scenario details match closely enough." The use of the terms "in general" and "closely enough" seems to imply that there are situations where the use of internal events and/or fire non-abandonment T/H runs are not appropriate. Therefore, in RAI 02.d (Reference 7), the NRC staff requested the licensee to provide a discussion of how the situations where the internal events and/or fire non-abandonment T/H runs are not appropriate were addressed. In response to RAI 02.d (Reference 3), the licensee explained that the thermal-hydraulic analysis for main control room accident sequences and MCRAB postulated sequences are not different for the fire PRA and that the differences are in the operator action response. The licensee explained that the fire HRA for MCRAB operator actions includes distinctions accounting for the differences in timing to accomplish credited operator actions for the differing scenarios (i.e., non-MCRAB and MCRAB). The licensee stated that the MCRAB credited operator actions include additional timing delays for the diagnosis, decision, and execution to establish the credited systems for MCRAB postulated scenarios based on procedures and operator interviews. These timing differences were addressed in the HRA. Because the licensee confirmed that scenario-specific HRA analyses were performed and justified, the NRC staff concludes this F&O has no impact on the application.

The disposition to open F&O 4-30 related to SR IGN-A7 states that the risk contribution from junction boxes has not been included in the fire PRA, but that the fire PRA will be modeled consistent with the guidance in Frequently Asked Question (FAQ) 13-0006, "Close-out of Fire Probabilistic Risk Assessment Frequently Asked Question 13-0006 on Modeling Junction Box Scenarios in a Fire PRA" (Reference 24), during the next PRA update. In RAI 02.e (Reference 7), the NRC staff requested the licensee to either provide justification that the risk from junction boxes has negligible impact on the 10 CFR 50.69 application, or to incorporate the risk associated with junction boxes into the fire PRA model. In response to RAI 02.e (Reference 3) and followup RAI 04.01 (Reference 4), the licensee proposed implementation item v to update the fire PRA model to model junction box fires consistent with FAQ 13-0006 prior to implementation of the 10 CFR 50.69 categorization process (see Section 3.5.5 of this SE).

F&O 4-34 related to SR FSS-G1 cited examples of transient fires that were excluded from consideration without justification. The disposition to this F&O stated that better documentation was needed for the basis of this screening but did not provide the basis for the exclusion. Therefore, in RAI 02.f (Reference 7), the NRC staff requested the licensee to identify the guidance used or explain why transient fires are excluded in specific scenarios such as those cited in the F&O, or alternatively, incorporate the excluded transient fires into the fire PRA model. In response to RAI 02.f (Reference 3), the licensee explained that this F&O is only related to transient fires in the multi-compartment analysis. In response to RAI 02.f and followup RAI 04.01 (Reference 4), the licensee proposed implementation item vi to update the fire PRA model to incorporate transient fires in the multi-compartment analysis, consistent with the accepted guidance in NUREG/CR-6850 (Reference 30), prior to implementation of the 10 CFR 50.69 categorization process (see Section 3.5.5 of this SE).

The disposition to F&O SR FSS-C6 explains that a focused-scope peer review was performed on the implementation of the thermally-induced electrical failure (THIEF) fire modelling tool, which resulted in two F&Os. In RAI 02.g (Reference 7), the NRC staff requested the licensee to either provide justification why the two F&Os have no impact on the 10 CFR 50.69 categorization, or to incorporate resolution to these two F&Os into the fire PRA model. In response to RAI 02.g (Reference 3), the licensee provided the two F&Os that resulted from the focused-scope peer review on the implementation of the THIEF fire modeling tool and summarized how they have been resolved. In disposition to one of the F&Os, FSS-H5-1, the licensee stated that the F&O has no impact on the 10 CFR 50.69 application because sensitivity studies were performed to address the uncertainties of the input parameters used for THIEF, and the results of sensitivity studies indicate that the THIEF parameter inputs have a negligible impact on the fire PRA. In followup RAI 02.02 (Reference 8), the NRC staff requested the licensee to summarize these sensitivity studies on the THIEF parameter inputs and provide their results demonstrating the negligible impact on the 10 CFR 50.69 categorization. In response to followup RAI 02.02 (Reference 4), the licensee explained that the THIEF model predicts the temperature profile within a cable as a function of time to simulate the delay in damage to a cable that allows additional time for manual suppression, thereby improving manual suppression probability. The licensee stated that it performed sensitivity studies on the THIEF calculation radial increment, the cable properties, and the conduit size to determine the impact on the manual suppression probability. Varying each of the parameters shows a relatively small impact on the manual suppression probability. The licensee noted that one of the sensitivity studies for fire PRA specified in Table 5-3 of NEI 00-04 removes all credit for manual suppression. The licensee proposed, through implementation item x, to conduct an additional sensitivity study during the categorization in which credit for immediate manual suppression will be assumed. The NRC staff finds that the addition of a sensitivity study is consistent with Table 5-3 in NEI 00-04, which states that additional sensitivity studies may be identified. The NRC staff has previously reviewed and validated the THIEF modelling technique (NUREG/CR-6931, Volume 3), and therefore, accepts the use of THIEF to support this application.

In RAI 03.a (Reference 7), the NRC staff identified that numerous changes had been made to the fire PRA after the peer review to resolve F&Os, including changes that appeared potentially significant based on dispositions presented in LAR Attachment 3, Table 3b to F&Os, which were considered resolved by the F&O closure review. NRC staff review of the reported changes indicated that some changes may use a new methodology and that the changes could impact significant accident sequences or the significant accident progression sequences (i.e., the change was an upgrade that should be peer reviewed). Accordingly, for those changes, the NRC staff requested clarification in RAI 03.a.i and justification of whether any of the changes made to resolve fire PRA F&Os met the definition of a PRA upgrade, which per the PRA standard would require a focused-scope peer review. These changes discussed in LAR Table 3b (i.e., resolved peer review findings) included:

- Changes to the fire PRA event trees to use the fire initiating event decision tree, as indicated in closed F&Os 2-12 and 4-2;
- Change to use of the "Fire Modeling Workbook" approach, as indicated in closed F&Os 2-24 and 4-17;
- Change to replace Fire Modelling Treatment notebook-based reduced heat release rates for transient fires of 60 kilowatts (kW) and 140 kW with guidance endorsed by the June 21, 2012, letter from Joseph Giitter to Biff Bradley entitled, "Recent Fire PRA Methods Review Panel Decisions and EPRI 1022993, 'Evaluation of Peak Heat Release Rates in Electrical Cabinets Fires'" (Reference 25), as indicated in open F&O 4-35;
- Change to use of the guidance in FAQ 14-0009, "Treatment of Well-Sealed MCC Electrical Panels Greater than 440V," dated October 20, 2014 (Reference 26), as indicated in closed F&O 4-26; and
- Change to use of the guidance in FAQ 12-0064, "Hot Work/Transient Fire Frequency: Influence Factors," dated September 5, 2012 (Reference 27), as indicated in closed F&O 5-7.

For each of the changes above, the licensee's response to RAI 03.a.i (Reference 3) summarized the original PRA model and the new PRA model and demonstrated that the changes extended methods previously used by the licensee to additional SSCs or scenarios. Therefore, none of the changes constitutes a PRA upgrade. In response to RAI 03.a.ii, the licensee identified two upgrades associated with the fire PRA that were identified in LAR Table 3a (i.e., open and partially resolved peer review findings). One upgrade was the implementation of the THIEF model. The licensee stated that this change was subject to a focused-scope peer review in June 2017. The F&Os from this peer review were provided by the licensee in response to RAI 02.g (Reference 3). The second upgrade identified by the licensee is a future action to model undesired operator actions in the fire PRA prior to implementation of the 10 CFR 50.69 categorization. The licensee included part of implementation item iv to model the undesired operator actions, conduct a focused-scope peer review, and resolve any F&Os prior to implementation of the 10 CFR 50.69 categorization process (see Section 3.5.5 of this SE).

Paragraph 50.69(c)(1)(i) of 10 CFR, requires, in part, that any plant-specific PRA used in the categorization must be of sufficient quality and level of detail to support the categorization process and must be subjected to a peer review process assessed against a standard that is endorsed by the NRC. RG 1.200 provides guidance for determining the technical adequacy of a fire PRA by comparing the PRA to the relevant parts of the ASME/ANS 2009 Standard using a peer review process. Based on its review, the NRC staff finds that the licensee has followed the guidance in RG 1.200 and submitted the results of the peer review, and therefore, meets the requirement in 10 CFR 50.69(b)(2)(iii). In the LAR supplement dated August 14, 2017, the licensee stated that the "Limerick Fire PRA uses methods that have been formally accepted by the NRC." The NRC staff has reviewed the peer review results and the licensee's resolution of the results and finds that the quality and level of detail of the fire PRA is sufficient to support the categorization of SSCs as required by 10 CFR 50.69 (b)(2)(ii) and use the process endorsed by the NRC staff in RG 1.201. Significant errors and weaknesses with the fire PRA will be resolved with the completion of implementation items iii, iv, v, vi (discussed in this section of the SE), and x (discussed in Section 3.5.2 of the SE). Therefore, the NRC staff concludes that the quality of the fire PRA with the completion of the implementation items iii, iv, v, vi, and x, meets the requirement in 10 CFR 50.69(c)(1)(i).

3.5.2 Importance Measures and Sensitivity Studies

Paragraph 50.69(c)(1)(i) of 10 CFR requires the results and insights from the PRA be used during categorization. These requirements are met, in part, by using importance measures and sensitivity studies, as described in the methodology in NEI 00-04, Section 5.0.

Fussell-Vesely and Risk Achievement Worth importance measures are obtained for each component and each PRA modeled hazard (i.e., separately for the internal events PRA and for the fire PRA) and the values are compared to specified criteria. Components that have internal event importance measure values exceeding the criteria are assigned HSS. Components that have fire event importance measures exceeding the criteria are assigned preliminary HSS. Integrated importance measures over all PRA modeled hazards are calculated per Section 5.6 of NEI 00-04, and components for which these measures exceed the criteria are assigned preliminary HSS.

The guidance in NEI 00-04 specifies sensitivity studies to be conducted for each PRA model. The sensitivity studies are performed to ensure that assumptions associated with these specific uncertain parameters (i.e., human error, common cause failure, and maintenance probabilities) are not masking the importance of a component. The NEI 00-04 guidance states that any additional "applicable sensitivity studies" from characterization of PRA adequacy should be considered. LAR Section 3.7 describes how the licensee searched for additional issues in the internal events (including internal flooding) PRA that should be evaluated with a sensitivity study. The licensee used the NRC guidance in NUREG-1855, "Guidelines on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making" (Reference 28), supplemented with the EPRI Technical Report (TR)-1016737, "Treatment of Parameter and Modeling Uncertainty for Probabilistic Risk Assessments" (Reference 29), to identify sources of uncertainty in the internal events PRA. Key assumptions documented in the licensee's PRA were also evaluated to identify additional sources of model uncertainty that could impact the 10 CFR 50.69 application. The assessment concluded that no additional sensitivity analyses were needed to address internal events PRA model-specific assumptions or sources of uncertainty.

In Table 2 of the LAR supplement dated August 14, 2017, the licensee provided the list of assumptions and sources of modelling uncertainty that were reviewed for the internal events (including internal flooding) PRA and the licensee's disposition. The NRC staff found that the dispositions for some of the assumptions and modeling uncertainties involve updating the PRA models prior to implementation of the 10 CFR 50.69 program. Accordingly, in RAI 04.a (Reference 7), the NRC staff requested the licensee develop an implementation item stating that 10 CFR 50.69 risk-informed categorization will not be performed until the PRA models are updated to address these uncertainties. In response to RAI 04 (Reference 3) and followup RAI 04.01 (Reference 4), the licensee proposed implementation items vii, viii, and ix, as follows:

vii. Update the pipe rupture frequencies in the internal flooding PRA to the most recent EPRI pipe rupture frequencies, as indicated on page 7 of supplement dated August 14, 2017.

viii. Remove credit for core melt arrest in-vessel at high reactor pressure vessel (RPV) pressure conditions from the internal events PRA model, as indicated on page 7 of supplement dated August 14, 2017.

ix. Update the PRA model to account for load shedding when crediting serial operation of high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) in loss of offsite power (LOOP) and station blackout (SBO) scenarios, as indicated on page 6 of the supplement dated August 14, 2017.

Given the licensee's assessment and its proposal to update the internal events PRA model before the 10 CFR 50.69 program is implemented, the NRC staff finds that the licensee searched for, identified, and evaluated sources of uncertainty in its internal PRA consistent with the guidance in NUREG-1855 and EPRI document TR-1016737, and therefore, satisfied the NEI 00-04 guidance to identify additional "applicable sensitivity studies."

LAR Section 3.7 describes how the licensee searched for additional assumptions and sources of uncertainties in the fire PRA. In Table 2 of the LAR supplement dated August 14, 2017, the licensee provided the list of assumptions and sources of modelling uncertainty that were reviewed for the fire PRA. Each of the 16 "tasks" in the NRC-endorsed methodology to perform fire PRA (NUREG/CR-6850) (supplemented by FAQs) is summarized in the LAR, and the sources of uncertainty in each task are described. The assessment concluded that, with the exception of implementation item x, no additional sensitivity analyses were needed to address fire PRA model-specific assumptions or sources of uncertainty.

In response to followup RAI 02.02 (Reference 4), and in implementation item x, the licensee identified that an additional sensitivity study will be performed during the categorization using the fire PRA in which immediate manual suppression will be credited for scenarios where manual suppression is modeled.

The NRC staff finds that the licensee searched for, identified, and evaluated sources of uncertainty in its fire PRA consistent with the relevant guidance in NUREG-1855 and EPRI document TR-1016737, and therefore, satisfied the NEI 00-04 guidance to identify additional "applicable sensitivity studies."

3.5.3 Non-PRA Methods

According to 10 CFR 50.69(c)(1)(ii), SSC functional importance must use an integrated, systematic process for addressing initiating events, SSCs, and plant operating modes, including those not modeled in the plant-specific PRA. The functions to be identified and considered include design-bases functions and functions credited for mitigation and prevention of severe accidents.

As described in the LAR, the licensee's categorization process uses the following non-PRA methods:

- SMA to assess seismic risk;
- Screening during the IPEEE to assess risk from other external hazards (high winds, external floods);
- Shutdown safety plan to assess shutdown risk.

The NRC staff's review of these methods is discussed below.

Seismic Risk

To assess seismic risk for the 10 CFR 50.69 categorization process, the licensee proposes to use the SMA method. SMA is a screening method that does not quantify CDF. The licensee used the EPRI SMA method described in EPRI NP-6041-SL, Revision 1, "A Methodology for Assessment of Nuclear Power Plant Seismic Margin" (Reference 31), during the IPEEE (U.S. Nuclear Regulatory Commission, "Individual Plant Examination of External Events (IPEEEs) for Severe Accident Vulnerabilities - 10 CFR 50.54(f)," Generic Letter 88-20, Supplement 4), dated June 1991 (Reference 32). The SMA method includes the development of the seismic safe shutdown equipment list (SSEL), which contains the components that would be needed during and after a seismic event. The SSEL identifies one preferred and one alternate path capable of achieving and maintaining safe shutdown conditions for at least 72 hours following an earthquake. The licensee stated in the LAR that it had updated the IPEEE SSEL to reflect the current as-built and as-operated plant. The licensee further stated that future changes to the plant will be evaluated as needed to determine their impact on the SMA and risk categorization process.

Consistent with NEI 00-04, the licensee's categorization process considers all components in the SSEL as HSS based on seismic risk. All components not listed in the SSEL are considered LSS with respect to seismic risk.

The method proposed by the licensee meets 10 CFR 50.69(c)(1)(ii) by using an integrated and systematic process to identify HSS components consistent with the seismic risk evaluation process, as described in the NRC-endorsed NEI 00-04. Therefore, the NRC staff finds the licensee's proposed method acceptable.

Other External Hazards (High Winds, External Floods)

As indicated in the LAR, external hazards were initially evaluated by the licensee during the IPEEE. This hazard category includes all non-seismic external hazards such as high winds, external floods, transportation and nearby facility accidents, and other hazards. The IPEEE external hazard analysis used a progressive screening approach and concluded that all these other hazards are negligible contributors to overall plant risk. Further, the licensee indicated that it had reevaluated these other external hazards using the criteria in the ASME/ANS 2009 Standard and screened all external hazards beyond seismic events. Therefore, the licensee proposed to treat all SSCs as LSS with respect to other external events risk. In response to RAI 06 (Reference 3), which requested additional information on the external event screening, the licensee stated that, "As part of the external hazard screening, an evaluation was performed to determine if there are components that participate in screened scenarios and whose failure would result in an unscreened scenario." This step is consistent with the process summarized in Figure 5-6 of NEI 00-04. The licensee stated that this process had been completed for flooding from local intense precipitation in an earlier flood hazard reevaluation report that was submitted to the NRC for review on March 12, 2015 (Reference 33), and the only HSS SSCs identified as part of this process were the diesel generator building exterior doors. The licensee stated that failure to credit the doors to the diesel generator building could result in an unscreened external flooding scenario, and therefore, the external doors would be HSS if the emergency diesel generator system were categorized.

In RAI 06.01.a (Reference 8), the NRC staff requested additional information from the licensee about the frequency of wind hazards that might cause damage to SSCs credited for safe shutdown. In the response to RAI 06.01.a (Reference 4), the licensee stated that the structures that directly affect the safe shutdown of the plant are designed to resist applicable design-basis tornado forces of 300 miles per hour that bound other winds. The licensee explained that some parts of the Emergency Service Water and Residual Heat Removal Service Water systems and spray pond networks, all associated with the ultimate heat sink, are not protected against design-basis wind and tornado missile hazards. The licensee referenced a 1984 hazard analysis that estimated the likelihood of the loss of the ultimate heat sink from tornado strikes as 8E-7/year. The licensee further clarified that the most recent estimates of frequency of wind speeds (see NUREG/CR-4461, "Tornado Climatology of the Contiguous United States" (Reference 34)), between 200 and 300 miles per hour has decreased from the 1984 estimates by factors between 50 and 10, and therefore, the damage frequency is at least a factor of 10 lower than 8E-7/year and can be quantitatively screened out. The licensee stated that the frequency of wind effect or wind-borne missiles to SSCs credited for safe shutdown is sufficiently low that the extreme wind or tornado hazard can be screened out without credit for mitigating SSCs. In response to RAI 06.01.c, the licensee further stated that the guidance in NEI 00-04, Section 5.4, is met. NEI 00-04, Section 5.4, includes guidance on evaluating the impact of all external hazard mitigating SSCs and identifying as HSS any SSCs whose failure would cause a screened hazard to become an unscreened hazard. The NRC staff finds that the licensee's previously accepted evaluation of the unprotected SSCs demonstrates that ultimate heat sink SSCs would not be HSS based on the frequency of tornado and tornado missiles, and the licensee has confirmed that NEI 00-04, Section 5.4, is met.

Because the licensee confirmed that the other external hazard risk evaluation is consistent with the NRC-endorsed NEI 00-04, the NRC staff finds the licensee's treatment of other external hazards acceptable, and 10 CFR 50.69(c)(1)(ii) is met.

Shutdown Risk

Consistent with the NEI 00-04 guidance endorsed by the NRC, the licensee proposes to use the shutdown safety assessment process based on NUMARC 91-06. NUMARC 91-06 provides considerations for maintaining DID for the five key safety functions during shutdown, namely, decay heat removal capability, inventory control, power availability, reactivity control, and containment - primary/secondary. NUMARC 91-06 specifies that a DID approach should be used with respect to each defined shutdown key safety function. This is accomplished by designating a running and an alternative system/train to accomplish the given key safety function.

In the licensee's response to RAI 07 (Reference 3), and consistent with the guidance in NEI 00-04, Section 5.5, the licensee indicated that components are categorized with respect to shutdown risk using a non-PRA shutdown assessment as follows:

- If a system/train supports a key safety function as the primary or first alternate means, then it is considered to be a "primary shutdown safety system" and is categorized as preliminary HSS. NEI 00-04 defines a "primary shutdown safety system" as also having the following attributes:
  - It has a technical basis for its ability to perform the function.
  - It has margin to fulfill the safety function.
  - It does not require extensive manual manipulation to fulfill its safety function.

- If the SSC's failure would initiate an event during shutdown plant conditions (e.g., loss of shutdown cooling, drain down), then that SSC is categorized as preliminary HSS.

As explained above, the shutdown safety assessment method proposed by the licensee is consistent with the guidance in NEI 00-04. In addition, the method meets 10 CFR 50.69(c)(1)(ii) by using an integrated and systematic process that could identify HSS components, if they existed, consistent with the shutdown evaluation process, as described in the NRC-endorsed NEI 00-04. Therefore, the NRC staff finds the licensee's proposed method acceptable.

3.5.4 Component Safety-Significance Assessment for Passive Components

Passive components are not modeled in the PRA, and therefore, a different assessment method is necessary to assess the safety-significance of these components. Passive components are those components having only a pressure-retaining function. This process also includes the passive function of active components such as the pressure/liquid retention of the body of a motor-operated valve.

In the LAR, the licensee proposed using a categorization method for passive components not cited in NEI 00-04 for passive component categorization, but approved by the NRC for Arkansas Nuclear One, Unit 2 (ANO-2) (Reference 35). The ANO-2 methodology is a risk-informed safety classification and treatment program for repair/replacement activities for Class 2 and Class 3 pressure-retaining items and their associated supports (exclusive of Class CC and MC items), using a modification of the ASME Code Case N-660, "Risk-Informed Safety Classification for Use in Risk-Informed Repair/Replacement Activities, Section XI, Division 1" (Reference 36). The ANO-2 methodology relies on the conditional core damage and large early release probabilities associated with pipe ruptures. Safety-significance is generally measured by the frequency and the consequence of, in this case, pipe ruptures. Treatment requirements (including repair/replacement) only affect the frequency of passive component failure. Categorizing solely based on consequences, which measures the safety-significance of the pipe given that it ruptures, is conservative compared to including the rupture frequency in the categorization. The categorization will not be affected by changes in frequency arising from changes to the treatment. Therefore, the NRC staff finds that the use of the repair/replacement methodology is acceptable and appropriate for passive component categorization of Class 2 and Class 3 SSCs.

In the LAR, the licensee did not specify what class of passive components will be categorized with the ANO-2 methodology. Therefore, in RAI 08 (Reference 7), the NRC staff requested the licensee to either confirm that only Class 2 and Class 3 SSCs will be categorized using ANO-2 passive methodology or to explain and justify how the methodology will be modified to include Class 1 components. In response to RAI 08 (Reference 3), the licensee stated its intent to apply the methodology to Class 1 pressure-retaining SSCs. The licensee further stated that this methodology does not require modifications in order to appropriately categorize Class 1 SSCs.

In followup RAI 08.01 (Reference 8), the NRC staff stated that since Class 1 SSCs constitute principal fission product barriers as part of the reactor coolant system or containment, the consequence of pressure boundary failure for Class 1 SSCs may be different than for Class 2 and Class 3, and therefore, the criteria in the ANO-2 methodology cannot automatically be generalized to Class 1 SSCs without further justification. A technical justification for Class 1 SSCs would have to address how the methodology is sufficiently robust to assess the safety-significance of Class 1 SSCs. This justification would have to include, but not be limited to, the following: (1) justification of the appropriateness of the conditional core damage probability numerical criteria used to assign 'high,' 'medium,' and 'low' safety-significance to these loss-of-coolant initiating events; (2) identification and justification of the adequacy of the additional qualitative considerations to assign 'medium' safety-significance (based on the conditional core damage probability) to 'high' safety-significance; (3) justification for crediting operator actions for success and failure of pressure boundary; and (4) guidelines and justification for selecting the appropriate break size (e.g., double-ended guillotine break or smaller break). The justification would also need to include supporting examples of types of Class 1 SSCs that would be assigned low safety-significance. The NRC staff found that the licensee's response to RAI 08 did not sufficiently justify how the ANO-2 methodology can be applied to Class 1 SSCs, and therefore, in followup RAI 08.01, the NRC staff requested the licensee to provide the requested technical justification or confirm the intent to apply the ANO-2 passive categorization methodology only to Class 2 and Class 3 SSCs.

In response to followup RAI 08.01 (Reference 4), the licensee stated that it will apply the process for the passive categorization of Class 2, Class 3, and non-Code class components, and that all ASME Code Class 1 SSCs with a pressure-retaining function, as well as supports, will be designated as HSS for the passive categorization. The licensee further clarified that this HSS designation for Class 1 SSCs cannot be changed by the IDP. Because all Class 1 SSCs and supports will be considered HSS, and only Class 2 and Class 3 SSCs will be categorized using the ANO-2 passive categorization methodology consistent with previous NRC staff approval, the NRC staff finds the licensee's proposed approach for passive categorization acceptable for the 10 CFR 50.69 categorization process.

3.5.5 Summary

The NRC staff reviewed the PRA and the non-PRA methods used by the licensee in its 10 CFR 50.69 categorization process to assess the safety-significance of active and passive components and finds these methods acceptable and consistent with RG 1.201 and the NRC-endorsed guidance in NEI 00-04. Accordingly, subject to the proposed license condition described below, the NRC staff approves the use of the following methods in the licensee's 10 CFR 50.69 categorization process:

- PRA to assess internal events, including internal flooding risk
- Fire PRA to assess fire risk
- SMA to assess seismic risk
- Screening using IPEEE to assess risk from other external hazards (high winds, external floods)
- Shutdown safety assessment process to assess shutdown risk
- ANO-2 (see Reference 35) passive categorization method to assess passive component risk for Class 2 and Class 3 SSCs and their associated supports

Based on its review of the LAR and the licensee's responses to the staff's RAIs, the NRC staff identified certain specific actions necessary to support their conclusion that the proposed program meets the requirements in 10 CFR 50.69 and the guidance in RG 1.201 and NEI 00-04. The licensee proposed the addition of a license condition for the implementation of 10 CFR 50.69 (see Section 4.0 of this SE). Specifically, the license condition states that Exelon will complete the implementation items listed in Attachment 2 of the Exelon letter to the NRC dated April 23, 2018 (Reference 4), prior to implementation of 10 CFR 50.69. Attachment 2 identifies nine implementation items that shall be completed prior to the implementation of the

10 CFR 50.69 categorization process, and one additional sensitivity study that will be completed as part of the categorization process:

i. Update the HRA pre-initiators in the internal events PRA model to meet Capability Category II of the ASME/ANS RA-Sa-2009 as endorsed by RG 1.200 Revision 2, conduct a focused-scope peer review of the pre-initiator analysis, and resolve any resulting F&Os, as indicated in response to RAI 01.a contained in Exelon letter dated January 19, 2018.

ii. Remove credit for recovery of instrument air from the internal events PRA model, as indicated in response to RAI 01.d contained in Exelon letter dated January 19, 2018.

iii. Update the success criteria for main steam isolation valve (MSIV) spurious opening, as indicated in response to RAI 02.a contained in Exelon letter dated January 19, 2018.

iv. Model undesired operator actions in the FPRA, conduct a focused-scope peer review, and resolve any F&Os, as indicated in response to RAI 02.c contained in Exelon letter dated January 19, 2018.

v. Update the FPRA model to model junction box fires consistent with frequently asked question (FAQ) 13-0006, as indicated in response to RAI 2.e contained in Exelon letter dated January 19, 2018.

vi. Update the FPRA model to incorporate transient fires in the multi-compartment analysis, as indicated in response to RAI 2.f contained in Exelon letter dated January 19, 2018.

vii. Update the pipe rupture frequencies in the internal flooding PRA to the most recent EPRI pipe rupture frequencies, as indicated on page 7 of supplement letter dated August 14, 2017.

viii. Remove credit for core melt arrest in-vessel at high reactor pressure vessel (RPV) pressure conditions from the internal events PRA model, as indicated on page 7 of supplement letter dated August 14, 2017.

ix. Update the PRA model to account for load shedding when crediting serial operation of high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) in loss of offsite power (LOOP) and station blackout (SBO) scenarios, as indicated on page 6 of the supplement letter dated August 14, 2017.

x. As part of the categorization process for the fire PRA, in addition to the list of fire PRA categorization sensitivities specified in NEI 00-04, Table 5-3, a sensitivity will be performed in which credit is taken for immediate manual suppression in scenarios in which manual suppression is already modeled, as indicated in Exelon letter dated April 23, 2018.

Additionally, the license condition states, in part, that prior NRC approval is required for a change to the categorization process that is specified in the license amendment and its supplements. The NRC staff's evaluation of the proposed license condition is in Section 4.0 of this SE.

3.6 Defense-in-Depth (NEI 00-04, Section 6)

NEI 00-04, Section 6.0, provides guidance on assessment of DID. Figure 6-1 in NEI 00-04 provides guidance to assess design-basis DID based on the likelihood of the design-basis internal event initiating event and the number of redundant and diverse trains nominally available to mitigate the initiating event. The likelihood of the initiating events is binned and, for different likelihood bins, HSS is assigned if fewer than the indicated number of mitigating trains are nominally available. Section 6 also provides guidance to assess containment DID based on preserving containment isolation and long-term containment integrity and on preventing containment bypass and early hydrogen burns. DID for beyond design-basis initiating events is addressed by the PRA categorization process.

RG 1.201 endorses the guidance in Section 6 but notes that the containment isolation criteria in this section of NEI 00-04 are separate and distinct from those set forth in 10 CFR 50.69(b)(1)(x). The criteria in 10 CFR 50.69(b)(1)(x) are to be used in determining which containment penetrations and valves may be exempted from the Type B and Type C leakage testing requirements in both Options A and B of Appendix J to 10 CFR Part 50, but the 10 CFR 50.69(b)(1)(x) criteria are not used to determine the proper RISC category for containment isolation valves or penetrations.

Section 6 indicates that the safety-significance determined by the guidance is HSS, and the licensee clarified in LAR Section 3.1.1 that it will require an SSC categorized as HSS based on the DID assessment in Section 6 to be categorized as HSS. Based on its review, the NRC staff finds the licensee's categorization process is consistent with the NRC-endorsed NEI 00-04 guidance and fulfills the 10 CFR 50.69(c)(1)(iii) criteria that DID is maintained.

3.7 Preliminary Engineering Categorization of Functions (NEI 00-04, Section 7)

All the information collected and evaluated in the different engineering evaluations is collected, organized, and provided to the IDP, as described in NEI 00-04, Section 7. The IDP will make the final decision about the safety-significance of SSCs based on guidelines in NEI 00-04, the information they receive, and their expertise.

In LAR Section 3.1.1, the licensee stated that if any component is identified as HSS from either the integrated PRA component safety-significance assessment (Section 5 of NEI 00-04) or the DID assessment (Section 6 of NEI 00-04), the associated system function(s) would be identified as HSS. Once a system function is identified as HSS, then all the components that support that function are categorized as preliminary HSS. In RAI 05.d (Reference 7), the NRC staff requested the licensee to clarify whether all aspects identified in Sections 5 and 6 of NEI 00-04, including if any components identified as HSS through Sections 5.3 to 5.5 of NEI 00-04 (dedicated to seismic, external hazards, or shutdown risk) will drive the system functions to be categorized as HSS. In response to RAI 05.d (Reference 3), the licensee explained that the safety-significance of functions will be categorized as preliminary HSS only if it is supported by a component determined to be HSS from a PRA-based assessment (i.e., for Limerick, internal events PRA and integrated PRA importance measures described in Section 5.6 of NEI 00-04). Components that are identified as HSS from using the non-PRA approaches (SMA, shutdown risk, other external hazards) will not drive the system function(s) they support to be assigned HSS. The licensee explained that non-PRA-based assessments result in the default categorization of any components associated with the safe shutdown success paths defined in those deterministic assessments to be HSS, regardless of its risk significance. The license referenced Section 7.1 of NEI 00-04, endorsed without comment in RG 1.201, which states:

> If any SSC is safety significant, from either the PRA-based component safety-significance assessment (Section 5) or the defense-in-depth assessment (Section 6), then the associated system function is preliminarily safety significant. All other functions/SSCs can be preliminarily assigned low safety-significance.

The NRC staff finds that the default assignment of LSS to functions associated with components that have been assigned HSS by non-PRA deterministic methods is consistent with NEI 00-04, and therefore, acceptable.

3.8 Risk Sensitivity Study (NEI 00-04, Section 8)

Paragraph 50.69(c)(1)(iv) of 10 CFR requires, in part, that any potential increases in CDF and LERF resulting from changes to treatment are small. The categorization process described in the NRC-endorsed NEI 00-04 guidance includes an overall risk sensitivity study for all the LSS components to confirm that if the unreliability of the components were increased, the increase in risk would be small (i.e., meet the acceptance guidelines of RG 1.174). LAR Sections 3.1.1 and 3.2.7 clarify that in the sensitivity study, the unreliability of all LSS SSCs modelled in the PRA(s) will be increased by a factor of 3. Separate sensitivity studies are to be performed for each system categorized, as well as a cumulative sensitivity study for all the SSCs categorized through the 10 CFR 50.69 process.

This sensitivity study, together with the periodic review process discussed in Section 3.11 of this SE, assure that the potential cumulative risk increase from the categorization is small. The performance monitoring process monitors the component performance to ensure that potential increases in failure rates of categorized components are detected and addressed before reaching the rate assumed in the sensitivity study. The NRC staff finds that the licensee will perform the risk sensitivity study consistent with the guidance in NEI 00-04, Section 8.0, and therefore, will assure that the potential cumulative risk increase from the categorization is small, as required by 10 CFR 50.69(c)(1)(iv).

3.9 Integrated Decisionmaking Panel Review and Approval (NEI 00-04, Sections 9 and 10)

Section 50.69(c)(2) of 10 CFR requires that the SSCs must be categorized by an IDP staffed with expert, plant-knowledgeable members whose expertise includes, at a minimum, PRA, safety analysis, plant operations, design engineering, and system engineering. LAR Section 3.1.1 clarifies that the IDP will be composed of a group of at least five experts who collectively have expertise in plant operation, design (mechanical and electrical) engineering, system engineering, safety analysis, and PRA. Therefore, the required expertise will be found in the IDP.

The guidance in NEI 00-04, endorsed in RG 1.201, provides confidence that the IDP expertise is sufficient to perform the categorization and that the results of the different evaluations (PRA and non-PRA) are used in an integrated, systematic process, as required by 10 CFR 50.69(c)(1)(ii). As provided by the NEI 00-04 guidance, and as indicated in LAR

Attachment 1, the process used by the IDP for the categorization of SSCs will be described and documented in a plant procedure.

LAR Section 3.1.1 states that at least three members of the IDP will have a minimum of 5 years of experience at the plant, and there will be at least one member of the IDP who has a minimum of 3 years of experience in modeling and updating of the plant-specific PRA. It further clarifies that the IDP will be trained in the specific technical aspects and requirements related to the categorization process. Training will address, at a minimum, the purpose of the categorization; present treatment requirements for SSCs, including requirements for design-basis events; PRA fundamentals; details of the plant-specific PRA, including the modeling, scope, and assumptions; the interpretation of risk importance measures, and the role of sensitivity studies and the change-in-risk evaluations; and the DID philosophy and requirements to maintain this philosophy.

Based on its review, the NRC staff finds that the licensee's IDP areas of expertise meet the requirements in 10 CFR 50.69(c)(2), and the additional descriptions of the IDP characteristics, training, processes, and decision guidelines are consistent with NEI 00-04, as endorsed by RG 1.201. Therefore, all aspects of the integrated, systematic process used to characterize SSCs will reasonably reflect current plant configuration and operating practices, and applicable plant and industry operational experience as required by 10 CFR 50.69(c)(1)(ii).

The licensee explained in response to RAI 05 (Reference 3) that the IDP's authority to change component categorization from preliminary HSS to LSS is limited. The licensee summarized these limitations in Table 1 of the response to RAI 05, as further revised in response to followup RAI 05.01 (Reference 4). As shown above in SE Table 1, and consistent with the guidance in NEI 00-04, components found to be HSS from the following aspects of the process cannot be re-categorized by the IDP:

- Internal events PRA (Section 5.1 of NEI 00.04),
- Integrated PRA component risk (Section 5.6 of NEI 00-04),
- SMA (Section 5.3 of NEI 00-04),
- Other external hazards (e.g., high winds, external floods (Section 5.4 of NEI 00-04)),
- Shutdown risk (Section 5.5 of NEI 00-04),
- DID (Section 6 of NEI 00-04), and
- Passive categorization.

Components categorized as HSS from either the fire PRA perspective or PRA sensitivity studies (for the internal events and the fire PRA), however, may be categorized as LSS by the IDP.

During its review, the NRC staff noted that Table 1 provided in the licensee's response to RAI 05 appeared to imply that the IDP could change HSS to LSS for the functions when using the qualitative criteria in NEI 00-04, Section 9.2. The licensee's proposal did not appear consistent with NEI 00-04 guidance. Therefore, in followup RAI 05.01 (Reference 8), the NRC staff requested the licensee to explain how the qualitative criteria in Section 9.2 will be used to determine the safety-significance of the system functions. In response to followup RAI 05.01 (Reference 4), the licensee explained that the assessments of the qualitative considerations are agreed upon by the IDP in accordance with Section 9.2. In some cases, a 10 CFR 50.69 categorization team may provide preliminary assessments of the seven considerations for the IDP's consideration; however, the final assessments of the seven considerations are the direct responsibility of the IDP. The licensee explained that the seven considerations are addressed preliminarily by the 10 CFR 50.69 categorization team. Each of the seven considerations requires a supporting justification for confirming (true response) or not confirming (false response) that consideration. If the 10 CFR 50.69 categorization team determines that one or more of the seven considerations cannot be confirmed, then that function is presented to the IDP as preliminary HSS. Conversely, if all seven considerations are confirmed, then the function is presented to the IDP as preliminary LSS. The IDP is responsible for reviewing the preliminary assessment at the same level of detail as the 10 CFR 50.69 team (i.e., all considerations for all functions are reviewed). The IDP may confirm the preliminary function risk and associated justification or may direct that it be changed based upon their expert knowledge. Because the qualitative criteria are the direct responsibility of the IDP, changes may be made from preliminary HSS to LSS or from preliminary LSS to HSS at the discretion of the IDP. However, the licensee stated that if the IDP determines that any one of the seven considerations cannot be confirmed (false response) for a function, then the final categorization of that function is HSS. Because the licensee explained that the consideration of the seven qualitative questions is the IDP's responsibility and that the final categorization of the function will be HSS when any one of the seven considerations cannot be confirmed (false response) for that function, the NRC staff finds the licensee's proposed use of the seven qualitative questions in the 10 CFR 50.69 categorization process acceptable and consistent with the guidance in NEI 00-04.

The IDP may change the categorization of a component from LSS to HSS based on its assessment and decisionmaking. As outlined in NEI 00-04, Section 10.2, and confirmed by the licensee in response to RAI 05 (Reference 3), the IDP may re-categorize components supporting an HSS function from HSS to LSS only if a credible failure of the component would not preclude the fulfillment of the HSS function and the component was not categorized as HSS based on the six criteria above (i.e., internal events PRA, integrated PRA component risk, SMA, shutdown, passive categorization, and DID). The licensee also explained that NEI 00-04, Section 4.0, discusses additional functions that may be identified (e.g., fill and drain) to group and consider potentially LSS components that may have been initially associated with an HSS function but that do not support the critical attributes of that HSS function.

Paragraph 50.69(c)(1)(iv) of 10 CFR requires, in part, reasonable confidence that sufficient safety margins are maintained for SSCs categorized as RISC-3. The licensee addresses safety margins through an integrated engineering evaluation that would nominally be addressed by the IDP. Consistent with the discussion in the NEI 00-04 guidance endorsed by RG 1.201, the IDP need not explicitly consider safety margins. Sufficient safety margin will be maintained because the RISC-3 SSCs will remain capable of performing their safety-related functions as required by 10 CFR 50.69(d)(2), and because any potential increases in CDF and LERF that might stem from changes in RISC–3 SSC reliability due to reduced treatment permitted by 10 CFR 50.69 will be maintained small, as required by 10 CFR 50.69(c)(1)(iv). Therefore, the NRC staff finds that the program implemented by the licensee, consistent with the endorsed guidance in NEI 00-04, fulfills the 10 CFR 50.69(c)(1)(iv) criteria that sufficient safety margins are maintained.

3.10 Program Documentation, Change Control, and Periodic Review (NEI 00-04, Sections 11 and 12)

Paragraph 50.69(c)(1)(ii) of 10 CFR requires, in part, that all aspects of the integrated, systematic process used to characterize SSC importance must reasonably reflect the current plant configuration and operating practices and applicable plant and industry operating experience. NEI 00-04, Section 11, provides guidance on program documentation and change control, and Section 12 provides guidance on periodic review. These sections are described in

NEI 00-04 with respect to satisfying 10 CFR 50.69(e) and 10 CFR 50.69(f), respectively. Maintaining change control and periodic review will also maintain confidence that all aspects of the program reflect current plant operation.

Section 50.69(e) of 10 CFR requires periodic updates to the licensee's PRA and SSC categorization. The NRC staff finds that changes over time to the PRA and SSC reliabilities are inevitable, and such changes are recognized by the 10 CFR 50.69(e) provision requiring periodic updates. As provided in RG 1.200, the NRC staff review of the PRA quality and level of detail reported in this SE is based primarily on determining how the licensee has resolved key assumptions and areas identified by peer reviewers as being of concern (i.e., F&Os). As discussed above in this SE, the NRC staff has concluded that several weaknesses or errors in the PRA will be addressed, as stated in the implementation items prior to implementation of the 10 CFR 50.69 categorization, because they otherwise could have a substantive impact on the PRA results. The results of the review of the current PRA are reported in Section 3.5 of this SE.

As described in the LAR Section 3.2.6, the licensee has administrative controls in place to ensure that the PRA models used to support the categorization reflect the as-built, as-operated plant over time. The licensee's process includes regularly scheduled and interim (as needed) PRA model updates. The process includes provisions for monitoring issues affecting the PRA models (e.g., due to changes in the plant, errors or limitations identified in the model, industry operational experience), for assessing the risk impact of unincorporated changes, and for controlling the model and associated computer files. The process also includes reevaluating previously categorized systems to ensure the continued validity of the categorization. Routine PRA updates are performed every two refueling cycles at a minimum. The NRC staff finds that this description is consistent with the requirements for feedback and process adjustment required by 10 CFR 50.69(e), and is, therefore, acceptable.

Section 50.69(f) of 10 CFR requires program documentation, change control, and records. In LAR Section 3.2.6, the licensee stated that it will implement a process that addresses the guidance in Section 11 of NEI 00-04 pertaining to program documentation and change control records. Section 3.1.1 of the LAR states that the RISC categorization process documentation will include the following ten elements:

- Program procedures used in the categorization
- System functions, identified and categorized with the associated bases
- Mapping of components to support function(s)
- PRA model results, including sensitivity studies
- Hazards analyses, as applicable
- Passive categorization results and bases
- Categorization results, including all associated bases and RISC classifications
- Component critical attributes for HSS SSCs
- Results of periodic reviews and SSC performance evaluations
- IDP meeting minutes and qualification/training records for the IDP members

In addition, LAR Attachment 1 (List of Categorization Prerequisites) states that the licensee will establish procedures prior to the use of the categorization process that will contain the following elements: (1) IDP member qualification requirements, (2) qualitative assessment of system functions, (3) component safety-significance assessment, (4) assessment of DID and safety margin, (5) review by the IDP and final determination of safety-significance for system functions and components, (6) risk sensitivity studies to confirm that the risk acceptance guidelines of

RG 1.174 are met, (7) periodic review to ensure continued categorization validity and acceptable performance for SSCs that have been categorized, and (8) documentation requirements identified in LAR Section 3.1.1. Procedures are formal plant documents, and changes will be tracked providing change control and records of the changes.

These categorization documents and records, as described by the licensee, include documentation and record change controls consistent with NEI 00-04, and endorsed by RG 1.201, and are in conformance with the requirements of 10 CFR 50.69(f)(1). Therefore, the NRC staff finds the documentation and records acceptable.

Based on its evaluation, the NRC staff finds that the change control and performance monitoring of categorized SSCs and PRA updates will sufficiently capture and evaluate component failures to identify significant changes in the failure probabilities. In addition, the PRA update program and associated reevaluation of component importance will appropriately consider the effects of changing failure probabilities and changing plant configuration on the component safety-significant categories. As discussed above, the staff finds the process in NEI 00-04 and the LAR will meet the requirements of 10 CFR 50.69(e) and 10 CFR 50.69(f), respectively. Therefore, the process used to characterize SSC importance will reasonably reflect the current plant configuration and operating practices, and applicable plant and industry operational experience required in 10 CFR 50.69(c)(1)(ii).

3.11 Technical Conclusion

The NRC staff reviewed the licensee's 10 CFR 50.69 categorization process and concludes that the licensee adequately implements 10 CFR 50.69 using models, methods, and approaches consistent with NEI 00-04, Revision 0, and RG 1.201, and therefore, satisfies the requirements of 10 CFR 50.69(c). Based on its review, the NRC staff finds the licensee's proposed categorization process acceptable for categorizing the safety significance of SSCs. Specifically, the staff concludes that the licensee's categorization process:

(1) considers results and insights from plant-specific internal events and fire PRAs that are of sufficient quality and level of detail to support the categorization process and that have been subjected to a peer review process against RG 1.200 Revision 2, as reviewed in Section 3.5.1 of this SE, and therefore, meets the requirements in 10 CFR 50.69(c)(1)(i);

(2) determines SSC functional importance using an integrated systematic process that reasonably reflects the current plant configuration, operating practices, and applicable plant and industry operational experience, as reviewed in Sections 3.3, 3.4, 3.5, 3.7, and 3.10 of this SE, and therefore meets the requirements in 10CFR 50.69(c)(1)(ii);

(3) maintains DID, as reviewed in Section 3.6 of this SE, and therefore, meets the requirements in 10CFR 50.69(c)(1)(iii);

(4) includes evaluations that provide reasonable confidence that for SSCs categorized as RISC-3, sufficient safety margins are maintained and that any potential increases in CDF and LERF resulting from changes in treatment are small, as reviewed in Sections 3.8 and 3.9 of this SE, and therefore, meets the requirements in 10 CFR 50.69(c)(1)(iv);

(5) is performed for entire systems and structures, rather than for selected components within a system or structure, as reviewed in Section 3.3 of this SE, and therefore, the requirements in 10CFR 50.69(c)(1)(v) will be met upon implementation; and
(6) includes categorization by IDP, staffed with expert, plant-knowledgeable members whose expertise includes, at a minimum, PRA, safety analysis, plant operation, design engineering and system engineering, as reviewed in Section 3.9 of this SE, and therefore, meets the requirements in 10CFR 50.69(c)(2).

4.0 10 CFR 50.69 IMPLEMENTATION LICENSE CONDITION

Section 50.69(b)(2) of 10 CFR requires the licensee to submit an application that describes the categorization process. Section 50.69(b)(3) of 10 CFR states that the Commission will approve the license application if it determines that the categorization process satisfies the requirements of 10 CFR 50.69(c). As described in this SE, the NRC staff has concluded that the 10 CFR 50.69 categorization process described in the licensee's application satisfies the requirements of 10 CFR 50.69(c). However, based on its review of the LAR and the licensee's responses to the NRC staff's RAIs, the NRC staff identified certain specific actions, as described below, that are necessary to support the staff's conclusion that the proposed program meets the requirements in 10 CFR 50.69 and the guidance in RG 1.201 and NEI 00-04.

The NRC staff's finding on the acceptability of the PRA evaluation in the licensee's proposed 10 CFR 50.69 process is conditioned on the completion of nine changes to the PRA and the addition of one sensitivity study to the studies summarized in Table 5-3 of NEI 00-04. These ten changes are identified as "Limerick 50.69 PRA Implementation Items" in Attachment 2 of the licensee's letter dated April 23, 2018. The staff notes that the licensee described some additional minor changes to the PRA and PRA methods. The staff determined that these minor changes would not impact the 10 CFR 50.59 categorization process and were similar to future changes to the PRA and PRA methods that occur over time. Therefore, the staff determined that these changes do not need to be resolved prior to implementation of the 10 CFR 50.69 process, and therefore, can be addressed and resolved using the licensee's periodic review process.

The licensee proposed the following condition to its license:

> 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 the license amendment No. [XX] (Unit X) and No. [XX] (Unit X), dated [DATE].
>
> 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).

Based on its evaluation in this SE, the NRC staff finds that the proposed license condition and its referenced implementation items are acceptable because they adequately implement 10 CFR 50.69 using models, methods, and approaches consistent with the applicable guidance that has previously been endorsed as acceptable by the NRC. For each implementation item, the licensee and the NRC staff have reached a satisfactory resolution involving the level of detail and main attributes that each remaining item will incorporate into the program upon its completion. Completion of these items does not change or impact the bases for the safety conclusions made by the NRC staff in the SE. The NRC staff, through an onsite audit or during future inspections, may choose to examine the closure of the implementation items with the expectation that any variations discovered during this review, or concerns regarding adequate completion of the implementation item, would be tracked and dispositioned appropriately under the licensee's corrective action program, and could be subject to appropriate NRC enforcement action, as completion of the implementation items would be required by the proposed license conditions.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the staff notified the Pennsylvania State official on June 12, 2018, of the proposed issuance of the amendments. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20 or change inspections or surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding, which was published in the *Federal Register* on September 26, 2017 (82 FR 44854), that the amendments involve no significant hazards consideration, and there has been no public comment on such finding. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

7.0 CONCLUSION

Based on the aforementioned considerations, the NRC staff has concluded that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that 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.

8.0 REFERENCES

1 Barstow J., Exelon Generation Company, LLC, letter to U. S. Nuclear Regulatory Commission, "Limerick Generating Station, Units 1 and 2, Renewed Facility Operating License Nos. NPF-39 and NPF-85, NRC Docket Nos. 50-353 and 50-353, Application to Adopt 10 CFR 50.69, Risk-Informed Categorization and Treatment of Stuctures, Systems, and Components (SSCs) for Nuclear Power Plants," dated June 28, 2017 (ADAMS Accession No. ML17179A161).

2 Barstow J., Exelon Generation Company, LLC, letter to U. S. Nuclear Regulatory Commission, "Limerick Generating Station, Units 1 and 2, Renewed Facility Operating License Nos. NPF-39 and NPF-85, NRC Docket Nos. 50-353 and 50-353, Supplement to Application to Adopt 10 CFR 50.69, Risk-informed Categorization and Treatment of SSCs for NPPs," dated August 14, 2017 (ADAMS Accession No. ML17226A336).

3 Barstow J., Exelon Generation Company, LLC, letter to U. S. Nuclear Regulatory Commission, "Limerick Generating Station, Units 1 and 2, Renewed Facility Operating License Nos. NPF-39 and NPF-85, NRC Docket Nos. 50-353 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," dated January 19, 2018 (ADAMS Accession No. ML18019A091).

4 Barstow J., Exelon Generation Company, LLC, letter to U. S. Nuclear Regulatory Commission, "Limerick Generating Station, Units 1 and 2, Renewed Facility Operating License Nos. NPF-39 and NPF-85, NRC Docket Nos. 50-353 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," dated April 23, 2018 (ADAMS Accession No. ML18113A870).

5 Barstow, J., Exelon Generation Company, LLC, letter to U.S. Nuclear Regulatory Commission, "Limerick Generating Station, Units 1 and 2, Renewed Facility Operating License Nos. NPF-39 and NPF-85, NRC Docket Nos. 50-353 and 50-353, Supplement to Application to Adopt 10 CFR 50.69, 'Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Plants'," dated July 27, 2018 (ADAMS Accession No. ML18208A179).

6 Sreenivas, V., U.S. Nuclear Regulatory Commission, letter to Hanson, Bryan, C., Exelon Nuclear, "Limerick Generating Station, Units 1 and 2 – Supplemental Information Needed for Acceptance of Requested Licensing Action Re: Adoption of Title 10 of the Code of Federal Regulations Section 50.69 (CAC Nos. MF9873 and MF9874)," dated July 31, 2017 (ADAMS Accession No. ML17207A077).

7 Sreenivas, V., U.S. Nuclear Regulatory Commission, E-Mail to Stewart, Glenn, Exelon Generation Company, LLC, "Limerick 50.69 License Amendment Request Application:

Request for Information (RAI)," dated December 6, 2017 (ADAMS Accession No. ML17341A250).

8 Sreenivas, V., U.S. Nuclear Regulatory Commission, E-Mail to Stewart, Glenn, and Helker, David, Exelon Generation Company, LLC, "Limerick Units 1 and 2: Second Request for Additional Information, Application to Adopt 10 CFR 50.69 Risk-Informed Categorization," dated March 26, 2018 (ADAMS Accession No. ML18085A643).

9 Nuclear Energy Institute, "10 CFR 50.69 SSC Categorization Guidline," NEI-00-04, dated July 2005 (ADAMS Accession No. ML052900163).

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

11 U.S. Nuclear Regulatory Commission, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance," Regulatory Guide 1.201 (For Trial Use), Revision 1, dated May 2006 (ADAMS Accession No. ML061090627).

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

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Date: July 31, 2018

B. Hanson

SUBJECT: 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) DATED JULY 31, 2018

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