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

October 25, 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: PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3 – ISSUANCE OF AMENDMENT NOS. 321 AND 324 TO ADOPT 10 CFR 50.69, "RISK-INFORMED CATEGORIZATION AND TREATMENT OF STRUCTURES, SYSTEMS AND COMPONENTS FOR NUCLEAR POWER REACTORS" (CAC NOS. MG0181 AND MG0182; EPID L-2017-LLA-0281)

Dear Mr. Hanson:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment Nos. 321 and 324 to Renewed Facility Operating License Nos. DPR-44 and DPR-56 for the Peach Bottom Atomic Power Station, Units 2 and 3, respectively, in response to your application dated August 30, 2017, as supplemented by letters dated October 24, 2017; May 7, 2018; June 6, 2018; August 10, 2018; and August 22, 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 the related safety evaluation is also enclosed. A notice of issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

Jenny Tobin

Jennifer Tobin, Project Manager
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-277 and 50-278

Enclosures:
1. Amendment No. 321 to Renewed DPR-44
2. Amendment No. 324 to Renewed DPR-56
3. Safety Evaluation

cc: Listserv

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

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-277

PEACH BOTTOM ATOMIC POWER STATION, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 321
Renewed License No. DPR-44

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 August 30, 2017, as supplemented by letters dated October 24, 2017; May 7, 2018; June 6, 2018; August 10, 2018; and August 22, 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. A new paragraph (17) is added to paragraph 2.C. of Renewed Facility Operating License No. DPR-44 to read as follows:

    (17) Adoption of 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems, and components for nuclear power plants"

    In support of implementing License Amendment No. 321 permitting the adoption of the provisions of 10 CFR 50.69 for Renewed Facility Operating License No. DPR-44 for Peach Bottom Unit 2, the license is amended to add the following license condition:

    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. 321 dated October 25, 2018.

    Exelon will complete the implementation items listed in Attachment 2 of Exelon letter to NRC dated June 6, 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).

4. 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: October 25, 2018

ATTACHMENT TO LICENSE AMENDMENT NO. 321

PEACH BOTTOM ATOMIC POWER STATION, UNIT 2

RENEWED FACILITY OPERATING LICENSE NO. DPR-44

DOCKET NO. 50-277

Add the following page to the Renewed Facility Operating License.

Remove	Insert
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(17) <u>Adoption of 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems, and components for nuclear power plants"</u>

In support of implementing License Amendment No. 321 permitting the adoption of the provisions of 10 CFR 50.69 for Renewed Facility Operating License No. DPR-44 for Peach Bottom Unit 2, the license is amended to add the following license condition:

(a) Exelon is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 structures, systems, and components (SSCs) using: Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events, including internal flooding, and internal fire; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 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. 321 dated October 25, 2018.

Exelon will complete the implementation items listed in Attachment 2 of Exelon's letter to the NRC dated June 6, 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. 50-278

PEACH BOTTOM ATOMIC POWER STATION, UNIT 3

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 324
Renewed License No. DPR-56

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 August 30, 2017, as supplemented by letters dated October 24, 2017; May 7, 2018; June 6, 2018; August 10, 2018; and August 22, 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. A new paragraph (17) is added to paragraph 2.C. of Renewed Facility Operating License No. DPR-56 to read as follows:

   (17) Adoption of 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems, and components for nuclear power plants"

   In support of implementing License Amendment No. 324 permitting the adoption of the provisions of 10 CFR 50.69 for Renewed Facility Operating License No. DPR-56 for Peach Bottom Unit 3, the license is amended to add the following license condition:

   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 3 License Amendment No. 324 dated October 25, 2018.

   Exelon letter to NRC dated June 6, 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).

4. 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: October 25, 2018

ATTACHMENT TO LICENSE AMENDMENT NO. 324

PEACH BOTTOM ATOMIC POWER STATION, UNIT 3

RENEWED FACILITY OPERATING LICENSE NO. DPR-56

DOCKET NO. 50-278

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

Remove	Insert
Page 7g	Page 7g
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(e) The results of the power ascension testing to verify the continued structural integrity of the steam dryer shall be submitted to the NRC staff in a report in accordance with 10 CFR 50.4. The report shall include a final load definition and stress report of the steam dryer, including the results of a complete re-analysis using the end-to-end B/Us from Peach Bottom Unit 2 benchmarking at EPU conditions. The report shall be submitted within 90 days of the completion of EPU power ascension testing for Peach Bottom Unit 3.

(f) During the first two scheduled refueling outages after reaching EPU conditions, a visual inspection shall be conducted of the steam dryer as described in the inspection guidelines contained in WCAP-17635-P.

(g) The results of the visual inspections of the steam dryer shall be submitted to the NRC staff in a report in accordance with 10 CFR 50.4. The report shall be submitted within 90 days following startup from each of the first two respective refueling outages.

(h) Within 6 months following completion of the second refueling outage, after the implementation of the EPU, the licensee shall submit a long-term steam dryer inspection plan based on industry operating experience along with the baseline inspection results.

The license condition described above shall expire: (1) upon satisfaction of the requirements in paragraphs (f) and (g), provided that a visual inspection of the steam dryer does not reveal any new unacceptable flaw(s) or unacceptable flaw growth that is due to fatigue, and; (2) upon satisfaction of the requirements specified in paragraph (h).

(16) <u>Maximum Extended Load Line Limit Analysis Plus (MELLLA+) Special Consideration</u>

The licensee shall not operate the facility within the MELLLA+ operating domain with a feedwater heater out of service resulting in more than a 10°F reduction in feedwater temperature below the design feedwater temperature.

(17) <u>Adoption of 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems, and components for nuclear power plants"</u>

In support of implementing License Amendment No. 321 permitting the adoption of the provisions of 10 CFR 50.69 for Renewed Facility Operating License No. DPR-44 for Peach Bottom Unit 2, the license is amended to add the following license condition:

(a) Exelon is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 structures, systems, and components (SSCs) using: Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events, including internal flooding, and internal fire; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 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 3 License Amendment No. 324 dated October 25, 2018.

Exelon will complete the implementation items listed in Attachment 2 of Exelon's letter to the NRC dated June 6, 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).

3. This renewed license is subject to the following conditions for the protection of the environment:

A. To the extent matters related to thermal discharges are treated therein, operation of Peach Bottom Atomic Power Station, Unit No. 3, will be governed by NPDES Permit No. PA 0009733, as now in effect and as hereafter amended. Questions pertaining to conformance thereto shall be referred to and shall be determined by the NPDES Permit issuing or enforcement authority, as appropriate .

B. In the event of any modification of the NPDES Permit related to thermal discharges or the establishment (or amendment) of alternative effluent limitations established pursuant to Section 316 of the Federal Water Pollution Control Act, the Exelon Generation Company shall inform the NRC and analyze any associated changes in or to the Station, its components, its operation or in the discharge of effluents therefrom. If such change would entail any modification to

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 321 TO

RENEWED FACILITY OPERATING LICENSE NO. DPR-44 AND

AMENDMENT NO. 324 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-56

EXELON GENERATION COMPANY, LLC

PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3

DOCKET NOS. 50-277 AND 50-278

1.0 INTRODUCTION

By letter dated August 30, 2017 (Reference 1), as supplemented by letters dated October 24, 2017 (Reference 2); May 7, 2018 (Reference 3); June 6, 2018 (Reference 4); August 10, 2018 (Reference 5); and August 22, 2018 (Reference 6), Exelon Generation Company, LLC (Exelon, the licensee) submitted a license amendment request (LAR) for the Peach Bottom Atomic Power Station, Units 2 and 3 (Peach Bottom or PBAPS). 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 [SSCs] for nuclear power reactors." The provisions of 10 CFR 50.69 allow adjustment of the scope of 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 October 10, 2017 (Reference 7), and e-mails dated April 6, 2018 (Reference 8), and July 10, 2018 (Reference 9), the U.S. Nuclear Regulatory Commission (NRC, the Commission) staff ("the staff") requested additional information from the licensee. By letters dated October 24, 2017; May 7, 2018; June 6, 2018; August 10, 2018; and August 22, 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 November 21, 2017 (82 FR 55404).

2.0 REGULATORY EVALUATION

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

The probabilistic approach to regulation is an extension and enhancement of traditional regulation by considering risk in a comprehensive manner. A probabilistic approach to regulation enhances and extends the traditional deterministic approach by allowing consideration of a broader set of potential challenges to safety, providing a logical means for prioritizing these challenges based on safety-significance, and allowing consideration of a broader set of resources to defend against these challenges. 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 published the 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 SSCs subject to special treatment requirements. Special treatment refers to those requirements that provide increased assurance beyond normal industry practices that SSCs can perform their design-basis functions. For SSCs determined to be of low safety-significance, alternative treatment requirements can be implemented in accordance with the regulation. For SSCs determined to be of high safety-significance, requirements are not changed. This approach allows improved focus on equipment that has high safety-significance, resulting in improved plant safety.

The rule (10 CFR 50.69) contains requirements on how a licensee categorizes SSCs using a risk-informed process, adjusts treatment requirements, consistent with the relative significance of the SSC, and manages the process over the lifetime of the plant. A risk-informed categorization process is employed to determine the safety-significance of SSCs and place the SSCs into one of four risk-informed safety class (RISC) categories. The determination of safety-significance is performed by an integrated decisionmaking process, as described by Nuclear Energy Institute (NEI) 00-04, Revision 0, "10 CFR 50.69 SSC Categorization Guideline" (Reference 10), 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.

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

2.2 Licensee's Proposed Changes

As provided in Attachment 2 of the licensee's letter dated August 10, 2018 (Reference 5), 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 Unit [2 or 3] License Amendment No. [321 or 324] dated October 25, 2018.

Exelon will complete the implementation items listed in Attachment 2 of Exelon letter to NRC dated June 6, 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).

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. This regulation permits power reactor licensees to implement an alternative regulatory framework with respect to special treatment. Section 50.69 of 10 CFR permits licensees to remove SSCs of low safety-significance from the scope of certain identified special treatment requirements and to revise requirements for SSCs of greater safety-significance. For SSCs determined to be of high safety-significance, requirements may not be changed.

Paragraph 50.69(b)(2) of 10 CFR states that the Commission will approve a licensee's implementation of this section by a license amendment if the Commission determines that the categorization process satisfies the requirements of 10 CFR 50.69(c). Paragraph 50.69(c) of 10 CFR requires licensees to use an integrated decisionmaking process to categorize safety-related and nonsafety-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):

RISC-1: Safety-related SSCs with safety-significant functions[1]
RISC-2: Nonsafety-related SSCs with safety-significant functions
RISC-3: Safety-related SSCs with low safety-significant functions
RISC-4: Nonsafety-related SSCs with 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). Licensees can then apply alternative treatments according to 10 CFR 50.69(b)(1) and 10 CFR 50.69(d), consistent with the categorization of the SSCs. 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. 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 release frequency (LERF) resulting from changes in treatment permitted by implementation of §§ 50.69(b)(1) and (d)(2) are small.

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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.

(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."

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) that imposes requirements to conform to Appendix B to 10 CFR Part 50, (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

The guidance in NEI 00-04, Revision 0 (Reference 10), 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 11), to address shutdown operations. The guidance in NEI 00-04 states that all SSCs relied on in the non-PRA approaches will be categorized as HSS, and therefore, the categorization is conservative.

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 11 provides specific guidance corresponding to 10 CFR 50.69(e).
- Section 12 provides specific guidance corresponding to 10 CFR 50.69(f).

NRC Regulatory Guide (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 12), 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, as 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. One acceptable approach to determining the technical adequacy of a PRA is contained in RG 1.200, "An Approach for Determining the Technical

Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities" (Reference 13). 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 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, 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") (Reference 14). The ASME/ANS 2009 Standard addresses internal events, fire, and other hazards. This RG provides guidance for determining the technical adequacy of a PRA by comparing the PRA to the relevant parts of 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 15), 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 of 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 NRC staff's review, and the documentation of that review in this SE, use 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. RG 1.201 also states that the implementation of all processes described in NEI 00-04 (i.e., Sections 2 through 12) is integral to providing reasonable confidence in the evaluations required by 10 CFR 50.69(c)(1)(iv) and 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).

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) 09.a

(Reference 8), 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.

In response to RAI 09.a (Reference 3), 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 09.a dated May 7, 2018, the process contains the following elements/steps:

- Defining system boundaries (System Engineering Assessment – Section 3.4 of this SE).

- Defining system functions and assigning components to functions (System Engineering Assessment – 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 (Assembly of Plant-Specific Inputs, Component Safety-significance Assessment – Sections 3.3 and 3.5 of this SE).

- Defense-in-depth (DID) characterization. (Defense-In-Depth Assessment – 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 3.4 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 (System Engineering Assessment Section 3.4 of this SE).

- Qualitative characterization. System functions are qualitatively categorized as HSS or LSS based on the seven questions in Section 9.2 of NEI 00-04 (IDP Review and Approval, 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 core damage frequency (CDF) and large early release frequency (LERF) and meets the acceptance guidelines of RG 1.174 (Risk Sensitivity Study, Section 3.8 of this SE).

- Review by the Integrated Decisionmaking Panel (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 (Preliminary Engineering Categorization of Functions, IDP Review and Approval, and SSC Categorization, Sections 3.7, 3.8, and 3.9 of this SE, respectively).

In response to RAI 09.b (Reference 3), 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 response to RAI 09.c (Reference 3), 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 09 and Table 1 of the RAI 09 response (Reference 3), 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, and therefore, acceptable to achieve reasonable confidence that the evaluations required by 10 CFR 50.69(c)(1)(iv) are performed.

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)

Section 3 of NEI 00-04 states that the assembly of plant-specific inputs involves the collection and assessment of the key inputs to the risk-informed categorization process. This includes design and licensing information, PRA analyses, and other relevant plant data sources. In addition, this step includes the critical evaluation of plant-specific risk information to ensure that they are adequate to support this application. The guidance in Section 3 of NEI 00-04 summarizes the use of risk information and the general quality measures that should be applied to the risk analyses supporting the 10 CFR 50.69 categorization as well as the characterization of technical acceptability of both the internal events at power PRA and other risk analyses necessary to implement 10 CFR 50.69.

The licensee's risk categorization process uses PRAs to assess risks from internal events (including internal flooding) and from fire. For the other applicable risk hazard groups, the licensee's process uses non-PRA methods for the risk characterization. The licensee uses its SMA to assess seismic risk, its Individual Plant Examination of External Events (IPEEE) Screening to assess the risk from other external hazards (high winds, external floods) and its Shutdown Safety Plan to assess shutdown risk. The use of risk information and quality of PRA is reviewed in Section 3.5 of this SE.

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 include design-basis functions and functions credited for mitigation and prevention of severe accidents. Section 4 of 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 assessment includes the following elements: system selection and system boundary definition, identification of system functions, and a mapping of components to functions.

[2] As further discussed in Section 3.9 of this SE, the licensee explained in response to RAI 09 that the seven qualitative criteria are assessed preliminarily 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.

Section 4 of NEI 00-04 states that system selection and boundary definition includes defining system boundaries where the system interfaces with other systems. Identification of system function includes the identification of all system functions, including design basis and beyond design-basis functions identified in the PRA, and making sure that system functions are consistent with the functions defined in design-basis documentation and Maintenance Rule functions. 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. In addition, Section 4 of NEI 00-04 states that the classification of SSCs having only a pressure-retaining function (also referred to as passive components), or the passive function of active components, should be performed using ASME Code Case N-660, "Risk-Informed Safety Classification for Use in Risk-Informed Repair/Replacement Activities" (Reference 30).

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).

In RAI 12 (Reference 8), the NRC staff requested explanation of how severe accident functions will be considered by the categorization process. In response to RAI 12 (Reference 3), the licensee explained that severe accident functions will be considered in the same manner that other components/functions are considered, as described in Section 3.2 of this SE. For the risk characterization elements (PRA modelled and non-modeled), identification of safety-significant SCCs or functions could lead directly to categorizing severe accident functions as safety-significant. The license explained that the core damage DID assessment element is limited to design-basis accidents, but the containment DID assessment element could identify safety-significant severe accident functions. The licensee explained that for application of the seven qualitative criteria listed in Section 9.2 of NEI 00-04, severe accident prevention or mitigation functions would not typically meet the criteria but if one did, then a severe accident function could be categorized as safety-significant. For passive categorization, the licensee explained that segments or components that support a severe accident function could be categorized as safety-significant.

In response to RAI 13 (Reference 3), the licensee confirmed that it will follow the guidance in NEI 00-04 that any functions/SSCs that serve as the interface between two or more systems will not be categorized until the categorization of all systems that they support is complete.

The process described in the LAR, as supplemented for categorizing components, is consistent with Section 4 of NEI 00-04 and capable of collecting and organizing information at the system level by defining boundaries, functions, and components, and therefore, the NRC staff finds that 10 CFR 50.69(c)(1)(ii) will be satisfied upon implementation. 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 because all system functions are identified and evaluated.

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

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. For the other risk hazards – 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, Seismic Margin Analysis, 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 are consistent with the methods included in the NEI 00-04 guidance, and therefore, acceptable to the NRC staff. The guidance considers the results and insights from the plant specific PRA as required by 10 CFR 50.69(c)(1)(i). 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. The licensee has had peer reviews of its internal events and fire PRAs. 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 therefore, the licensee has satisfied the requirements that the PRA be subjected to a peer review process and that the results of that process be submitted in the application.

Internal Events PRA

The NRC staff reviewed the results of the peer review of the internal events and internal flooding PRA and associated facts and observations (F&O) closure review described in LAR Sections 3.2.1 and 3.3 and Attachment 3. As clarified by the licensee in response to RAI 01 (Reference 3), the last full-scope peer review of the internal events PRAs (including internal flooding) was performed in 2010 against PRA Standard ASME/ANS RA-Sa-2009 (Reference 14), as endorsed by RG 1.200, Revision 2, using the NEI 05-04 process (Reference 17).

As stated in the LAR, an independent assessment (IA) F&O closure review was performed in November 2016 by an independent assessment team for the internal events, internal flooding, and fire finding level F&Os. This November 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 17), NEI 07-12 (Reference 19), and NEI 12-13, "Close Out of Facts and Observations" (Reference 20). The NRC staff accepted, with conditions, a final version of Appendix X in the letter dated May 3, 2017 (Reference 21), which differed from the guidance used by the licensee in the November 2016 F&O closure.

Because this IA F&O closure review was performed prior to the NRC acceptance of the IA F&O closure process, the NRC staff requested the licensee to explain how the IA F&O closure was consistent with the process documented in Appendix X (Reference 18), as accepted by NRC in the staff memorandum dated May 3, 2017 (Reference 21). In the supplement dated October 24, 2017 (Reference 2), the licensee stated that the independent assessment team retrospectively addressed the differences between the guidance used and the approved May 3, 2017, version of Appendix X, and issued a revision to the F&O finding closure technical review report. Specifically, the licensee clarified that the evaluation and the final F&O finding closure technical report was revised to include or confirm the inclusion of the basis for whether or not each finding resolution represented PRA maintenance or PRA upgrade; that no newly developed methods were reviewed; and that the aspects of the underlying surveillance requirements that were previously not met, or met at Capability Category (CC) I, are now met, or met, at CC II. For PRA standards, Capability Categories I and II are defined in the ASME/ANS guidance (Reference 14) and are based on site specificity and model realism. The NRC staff finds that the licensee addressed all the differences between the guidance used for the November 2016, IA F&O closure evaluation and the final guidance, and therefore, accepts that the findings were reviewed and closed using the process documented in Appendix X, as accepted by NRC in the staff memorandum dated May 3, 2017.

In LAR Attachment 3, the licensee submitted the F&Os that were not closed by the IA F&O closure review. For each F&O, the licensee provided a disposition for this application. In response to RAI 14 (Reference 3), the licensee provided three additional open F&Os that were not provided in the LAR. The licensee explained that these findings were internally identified as open and have not been provided to the F&O closure review team.

The NRC staff reviewed the licensee's resolution of all the peer review findings and assessed 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 one of the findings, as described in the following paragraph.

F&O 2011-3-1 (and associated F&Os 2011-3-4, 2011-3-6, and 2011-5-8) found that the test and maintenance pre-initiators were not derived from a review of procedures and practices, as specified in Supporting Requirement HR-A1 of the PRA Standard. In RAI 03.a (Reference 8), the NRC staff requested that the licensee justify its statement that not resolving the F&Os has a minimal impact on the application, or to provide a mechanism that ensures a review of procedures and practices at the plant is conducted, and that any pre-initiators identified from the review are included in the PRA models prior to implementing the 10 CFR 50.69 categorization process. In response to RAI 03.a (Reference 3), the licensee proposed implementation item #1 to update the human reliability analysis (HRA) pre-initiators in the internal events PRA model to meet CC II of the PRA Standard, to conduct a focused-scope peer review of the pre-initiators analysis, and to resolve any F&Os resulting from the focused-scope peer review prior to implementation of the 10 CFR 50.69 categorization 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, met the requirement in 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 licensee's internal events and internal flooding 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 and internal flooding PRA will be resolved prior to implementation of the 10 CFR 50.69 categorization process with the completion of implementation item #1 (discussed in this Section of this SE) and implementation items #9, #10, and #11 (discussed in Section 3.5.1 of this SE). Therefore, the NRC staff concludes that the licensee has submitted the results of the PRA review process for its internal events and internal flooding PRA, as required by 10 CFR 50.69(b)(2)(iii), and that the internal events and internal flooding PRA, with the completion of the proposed implementation items #1, #9, #10, and #11, meet the requirement in 10 CFR 50.69(c)(1)(i).

Fire PRA

In the LAR, the licensee stated that the "internal Fire PRA model was developed, consistent with NUREG/CR-6850, and only utilizes methods previously accepted by the NRC." 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 Attachment 3. The licensee's response to RAI 01 (Reference 3) clarifies that the fire PRA was subject to a full-scope industry peer review in December 2012 against PRA Standard ASME/ANS RA-Sa-2009 (Reference 14), as endorsed by RG 1.200, Revision 2 (Reference 16), using the NEI 07-12 process (Reference 19).

In November 2016, an IA F&Os closure review was performed by an independent assessment team on fire events finding level F&Os. The IA F&O closure review is discussed above in Section 3.3.1 of this SE.

In the LAR supplement dated October 24, 2017, the licensee stated that one PRA upgrade was identified during the F&O closure review. In response to RAI 02 (Reference 3), the licensee explained that the cited upgrade was regarding the use of the following three models: (1) thermally-induced electrical failure (THIEF), 2) flame spread over horizontal cable (FLASHCAT), and 3) time-to-automatic-detection calculations. The licensee stated that a focused-scope peer review was performed on this upgrade, concurrent with the 2016 F&O closure process. It further clarified that the resulting F&Os were included in LAR Attachment 3.

The NRC staff reviewed the licensee's resolutions for all the fire PRA peer review findings provided by the licensee 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.

The disposition to F&O 2012-1-33 resolved a number of concerns associated with transient fire area weighting factors, except for the need to address obstructed floor area. In RAI 03.b, the NRC staff requested that the licensee justify its statement that not resolving the F&O has a minimal impact on the application or to provide a mechanism that ensures that the floor area ratios are adjusted for obstructed floor area in the fire PRA model prior to implementing the 10 CFR 50.69 categorization process. In response to RAI 03.b (Reference 3), the licensee proposed implementation item #2 to adjust the transient floor area ratios to consider obstructed floor space, in order to provide a more accurate distribution of transient ignition frequency, prior to implementation of the 10 CFR 50.69 process.

The disposition to F&O 2012-1-40 states that detailed two-point fire modeling was not performed for all risk significant scenarios, but that this treatment did not have a significant impact on the application. In RAI 03.c (Reference 8), the NRC staff requested that the licensee justify its statement that not fully resolving the F&O has a minimal impact on the application or to provide a mechanism that ensures that the two-point fire modeling is applied to risk-significant fire scenarios prior to implementing the 10 CFR 50.69 categorization process. In response to RAI 03.c (Reference 3), the licensee proposed implementation item #3 to use the two-point fire modelling for scenarios "capable of being modelled with a two-point fire modeling approach" prior to implementation of the 10 CFR 50.69 process. In RAI 03.c.01, the NRC staff requested the licensee to explain what types of scenarios would not be modeled with this approach. In response to RAI 03.c.01, the licensee clarified that, at minimum, a two-point fire intensity model will be used in accordance with guidance from NUREG/CR-6850 (Reference 22) and that the only exceptions will be for using other NRC-approved guidance. The licensee cited the use of fire PRA Frequently Asked Question 13-0005 (Reference 23) for special case cable fires, such as self-ignited cable fires, and Frequently Asked Question 13-0006 (Reference 24) for junction box fires.

The disposition to F&O 2012-3-17 states that a review of potentially vulnerable fire wrap configurations is performed and that credit for configurations confirmed to be susceptible to mechanical damage will be removed from the fire PRA. In RAI 03.d (Reference 8), the NRC staff requested that the licensee justify its statement that not fully resolving the F&O has a minimal impact on the application or to provide a mechanism that ensures that fire wrap subject to mechanical damage is not credited in fire scenarios. In response to RAI 03.d (Reference 3), the licensee proposed implementation item #4 to perform a review of potentially vulnerable fire wrap configurations and to remove credit from the fire PRA for fire wrap susceptible to mechanical damage.

F&O 2012-3-37 states that not all modeled, non-propagating electrical fires considered failure of the panel, and targets terminating at the panel in the fire PRA. The disposition stated that the licensee will confirm that the excluded panels lead to a single failure, and that the licensee will incorporate any excluded panels into the fire PRA if the licensee determines that multiple failures from an excluded panel fire are possible. In RAI 03.e (Reference 8), the NRC staff requested how the resolution to F&O 2012-3-37 was consistent with guidance in NUREG/CR-6850 (Reference 22). In response to RAI 03.e (Reference 3), the licensee stated that, consistent with guidance in NUREG/CR-6850, the excluded electrical panels were excluded because they are simple wall-mounted panels with less than four switches or they are well-sealed and robustly secured cabinets containing circuits below 440 volt (V). Accordingly, no update to the fire PRA is needed to resolve F&O 2012-3-37. Because the licensee's exclusion of electrical panels is in alignment with guidance for fire PRA in NUREG/CR-6850, the NRC staff concludes this F&O has no impact on the application.

The disposition to F&O 2012-6-1 explained that two internal events F&Os remain open. In RAI 03.f (Reference 8), the NRC staff noted that one of these two F&Os (i.e., F&O 3-6) is already addressed in RAI 03.a, but that the other F&O (i.e., F&O 6-11) pertaining to lack of validation for use of Maintenance Rule unavailability data in the PRA models, is not resolved for the application. In response to RAI 03.f (Reference 3), the licensee explained that the raw data taken directly from the Maintenance Rule provides realistic testing and failure probabilities, because any testing or maintenance that would not make the SSC unavailable would not be tracked in the Maintenance Rule. The licensee explained that the plant procedures stipulate that out-of-service SSCs are considered unavailable for the Maintenance Rule unless their function can be promptly restored by an operator. These restoration actions must be uncomplicated, proceduralized, and must not require diagnosis or repair. Therefore, the licensee concluded that the resolution of this F&O is a documentation issue and will have no impact on the internal events or fire PRA. Because the licensee justified the use in the PRA model of the raw unavailability data taken directly from the Maintenance Rule, the NRC staff concludes that the resolution of internal events F&O 6-11 cited in FPRA F&O 2012-6-1 is acceptable for this application.

F&Os 2012-5-6, 2016-1-1, 2016-1-2, and 2016-1-6 concern uncertainty associated with modeling inputs used in THIEF and FLASHCAT. The outputs from THIEF and FLASHCAT are used to determine the time to detection for electric cable fires, which in turn is used to determine non-suppression probabilities, using the guidance from NUREG/CR-6850. The dispositions to each of these F&Os explain that sensitivity studies were performed to determine the impact of input parameter assumptions. In RAI 04 (Reference 8), the NRC staff requested that the licensee provide the results of the sensitivity studies demonstrating that the cited assumptions have no impact on the application. In response to RAI 04 (Reference 3), the licensee proposed implementation item #5 to perform a sensitivity study during the 10 CFR 50.69 categorization process, in which immediate manual suppression is assumed. This sensitivity will be in addition to the sensitivity study already specified in NEI 00-04, Table 5-3, which assumes no credit for manual suppression. The licensee stated that the two sensitivity studies together bound the impact of non-suppression determined by the cited fire modeling concerns. Accordingly, the NRC staff finds that the uncertainty cited in F&O 2016-1-1 is addressed by implementation item #5 to perform the sensitivity studies during the 10 CFR 50.69 categorization. Additionally, in response to RAI 04.a and RAI 04.d, the licensee stated that uncertainty associated with using generic weighted average input parameters for FLASHCAT will be replaced with scenario and cable-specific parameters (mass per unit length and plastic mass fraction) for fires modelled in the fire PRA. The licensee proposed implementation item #6 to perform this parameter update prior to implementation of the 10 CFR 50.69 categorization process.

The dispositions to F&O 2012-2-6 and F&O 2012-2-7 state that the licensee will perform a procedure-by-procedure review to determine if fire-induced instrumentation damage can produce misleading indication, leading to undesired operator actions. The disposition also states that if such possibilities are identified, then the undesired operator actions will be modeled in the fire PRA. In RAI 05.a (Reference 8), the NRC staff requested description of how modeling of undesired operator actions will be performed and proposal of a mechanism to ensure a focused-scope peer review is performed if such modeling is determined to constitute a PRA upgrade as defined by the PRA Standard. In response to RAI 05.a (Reference 3), the licensee stated that the guidance in NUREG-1921 (Reference 25) specific to modeling undesired operator actions will be used to incorporate these events. The licensee provided implementation item #7 to use the guidance in NUREG-1921 for identifying and modeling undesired operator actions, conduct a focused-scope peer review of the upgrade, and resolve any resulting new F&Os prior to implementation of 10 CFR 50.69 categorization.

The disposition to F&O 2012-5-1 states that if the licensee identifies uncoordinated circuits, then the uncoordinated circuits will be modeled in the fire PRA. In RAI 05.b (Reference 8), the NRC staff requested description of how modeling of uncoordinated circuits will be performed and to provide a mechanism to ensure a focused-scope peer review is performed if such modeling is determined to constitute a PRA upgrade as defined by the PRA Standard. In response to RAI 05.b (Reference 3), the licensee stated that if uncoordinated circuits are identified, then the uncoordinated circuits will be incorporated into the fire PRA. For those circuits, the model will fail the power supply associated with an upstream breaker that can open due to the lack of coordination. The licensee explained that this modeling update is not considered a PRA upgrade because it does not require modeling additional components or component failure modes, but rather modelling failure of the power supply as defined above. In response to RAI 05.b, the licensee proposed implementation item #8 to perform this modeling for circuits identified to be uncoordinated. In response to RAI 05.b.01 (Reference 4), the licensee clarified that the evaluation needed to show that circuits are coordinated will be performed in accordance with the guidance in NUREG/CR-6850 on breaker coordination studies. The NRC staff finds the licensee's response acceptable because the licensee will resolve the F&O using approved NRC guidance.

In the dispositions to F&O 2016-1-3, F&O 2016-1-4, and F&O 2016-1-9 resulting from the focused-scope peer review, the licensee confirmed that THIEF, FLASHCAT, and time-to-automatic-detection model parameters are used within their limits of applicability. In response to RAI 06 (Reference 3), the licensee addressed the adequacy of the modeling for the application for each of the three F&Os cited in the RAI.

For F&O 2016-1-3 concerning lack of demonstration that THIEF was used within its validation range, the licensee quoted from NUREG/CR-6931 (Reference 26). The guidance in NUREG/CR-6931, Volume 3, on THIEF modeling states that THIEF is applicable to a wide variety of cables with no need for additional information beyond the cable diameter, mass per length, and an empirical failure temperature. The licensee further states that NUREG-1805, Supplement 1 (Reference 27), includes a THIEF model, which considers cable parameters outside of the parameters of the cables tested in NUREG/CR-6931.

For F&O 2016-1-4 concerning the need to show that the time-to-automatic-detection model was used within the known limits of its applicability, the licensee stated that it followed the reviewer's recommendation to add the ceiling jet distance ratio to the set of non-dimensional parameters to ensure the correlation is used within its known limits for each scenario. The licensee stated, however, that there are cases in which the ratio would be outside the validation range presented in NUREG-1824 (Reference 28) on verification and validation of fire models. However, as explained earlier in this SE, the THIEF/FLASHCAT/time-to-detection modeling that the categorization process defined by NEI 00-04, Table 5-3, includes a sensitivity study, which removes credit for manual suppression and, in addition, the licensee proposed implementation item #5 to add another sensitivity study to be performed during the categorization process in which immediate manual suppression is credited. These sensitivity studies bound the range of non-suppression credit that can be determined by the modeling.

For F&O 2016-1-9 concerning verification that FLASHCAT and THIEF were properly incorporated into the Microsoft Excel spreadsheets, the licensee stated that implementation of THIEF and FLASHCAT was verified by comparing results of the implemented models with other fire models such as Fire Dynamic Tools, Electric Power Research Institute (EPRI) Fire-Induced Vulnerability Evaluation (FIVE) (Reference 29), and Microsoft Excel Fire Modeling Workbook cases. The licensee stated that this comparison study demonstrates that the models were implemented appropriately. Because of efforts to demonstrate that the THIEF, FLASHCAT, and time-to-detection modeling was performed within their limits of applicability, and because of the sensitivity studies that will be performed to bound the impact that this modeling could have on the non-suppression probabilities used on the fire PRA, the NRC staff concludes that the resolutions for F&Os 2012-1-1, F&O 2012-1-4, and F&O 2012-1-9 are adequate for this application.

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). 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 using the process endorsed by the NRC staff in RG 1.201. Significant errors and weaknesses in the fire PRA will be resolved with the completion of implementation items #2, #3, #4, #5, #7, and #8 (discussed in this section of this SE) and implementation item #6 (discussed in Section 3.5.1 of this SE). Therefore, the NRC staff concludes that the licensee has submitted the results of the PRA review process for its fire PRA, as required by 10 CFR 50.69(b)(2)(iii), and that the quality of the fire PRA with the completion of the implementation items #2, #3, #4, #5, #6, #7, and #8, meet the requirement in 10 CFR 50.69(c)(1)(i).

3.5.2 Important Measures and Sensitivity Studies

Paragraph 50.69(c)(1)(i) of 10 CFR requires the licensee to consider the results and insights from the PRA 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.

Fussell-Vesely (F-V) and Risk Achievement Worth (RAW) 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 in NEI 00-04. Components that have internal event importance measure values that exceed the criteria are assigned HSS and cannot be changed by the IDP. 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 uncertainty 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 3.2.7 describes how the licensee searched for additional issues in the internal events (including internal flooding) and fire PRAs 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 32), supplemented with the EPRI Technical Report (TR)-1016737, "Treatment of Parameter and Modeling Uncertainty for Probabilistic Risk Assessments" (Reference 33), to identify sources of uncertainty in the internal events PRA.

Attachment 6 of the LAR includes dispositions to a number of key assumptions and sources of modeling uncertainty that in some cases did not provide sufficient information for the NRC staff to conclude that the uncertainty did not impact the application. Therefore, the NRC staff requested additional information to clarify the licensee's dispositions as described in the following paragraphs. As discussed below, the NRC staff found that some assumptions were adequately resolved and three required future updating of the PRA models prior to implementation of the 10 CFR 50.69 program and are implementation items (i.e., regulatory obligations). The licensee proposed, and the staff accepted, one regulatory commitment, consistent with the NRC staff's significant interest in the assigned failure probability in one assumption for which the licensee demonstrated an alternative conservative assumption that had a known and acceptable impact on the categorization results.

The NRC staff found that the dispositions for some of the assumptions and modeling uncertainties involved updating the PRA models prior to implementation of the 10 CFR 50.69 program. Accordingly, in RAI 07, the NRC staff requested the licensee to further justify these uncertainties. In response to RAI 07.a (Reference 3), the licensee proposed implementation items #10 and #11 as follows:

10. The pipe rupture frequencies will be updated in the internal flooding PRA to the most recent EPRI pipe rupture frequencies.

11. Credit for core melt arrest in-vessel at high reactor pressure vessel (RPV) pressure conditions will be removed from the internal events PRA model.

In Attachment 6 to the LAR, the licensee stated that when the outdoor air temperature exceeds the design-basis temperature for the diesel generators (DGs) (i.e., > 80 degrees Fahrenheit (°F)), then two DG cooling fans are required. However, the licensee stated that it assumes just 1-of-2 DG cooling fans to be an adequate success criterion for the entire year. In RAI 08.a (Reference 8), the NRC staff requested that the licensee provide the basis for the adequacy of the 1-of-2 DG cooling fan success criteria or to provide a mechanism that ensures that the PRA models are adjusted to account for higher summer temperatures prior to implementing the 10 CFR 50.69 categorization process. In response to RAI 08.a (Reference 3), the licensee stated that site weather information will be obtained to determine the amount of time in recent history that the outdoor air temperature exceeded the design temperature for the DGs, and that the PRA models will be adjusted to reflect a success criterion of 2-of-2 DG cooling fans for those periods of time. The licensee proposed implementation item #9 to update the PRA models with adjusted DG cooling fan success criteria to account for the period of time the outdoor temperature is above the design temperature of the DGs.

The licensee stated that the basis for crediting low pressure injection after the core damage to avoid large early release is "reasonable best-estimate approach," which has a minor impact on the application. In RAI 08.b, the NRC staff requested the licensee to justify this assumption and to explain the term "timely low pressure injection." In response to RAI 08.b (Reference 3), the licensee stated that thermohydraulic basis was established through a sensitivity study performed using MAAP4 (Reference 34). The licensee stated that the MAAP4 results showed that emergency depressurization and restoration of one train of low pressure core injection after core damage, but before vessel failure, averts vessel failure. Based on its analysis, the licensee defined timely injection as at least 30 minutes before vessel failure. Because the licensee established a thermohydraulic basis for use of timely low pressure injection to avert large early release, the NRC staff concludes that this uncertainty is adequately addressed for this application.

The licensee stated that although the safety relief valves (SRVs) are not tested to perform their function following the passing of liquid, a nominal failure probability was assigned to the failure of the SRVs to successfully open after flooding of the steam lines. In response to RAI 08.c.i (Reference 3), the licensee stated that the SRV failure to open is consistent with the data in NUREG/CR-6928 (Reference 35). In RAI 08.c, the NRC staff also requested justification for the probability assigned to the failure of the SRV to close after passing liquid. The licensee explained in response to RAI 08.c.ii that this failure mode is not modeled in the PRA because it leads to depressurization, which is a desired state, allowing the low pressure systems to inject. The NRC staff concludes that this uncertainty is adequately modeled for this application because the licensee's assumed failure probability for SRVs to open is consistent with published industry average data, and because the licensee justified not modeling the SRV failure to close in the PRA model.

The licensee stated that the reactor core isolation cooling (RCIC) and high-pressure coolant injection (HPCI) turbines were assumed to continue to function after ingesting liquid instead of steam. The licensee stated that the failure of the turbines to run was assigned a nominal failure probability. In RAI 08.d, the NRC staff requested clarification of whether the turbines were designed to continue running after ingesting liquid and explanation and justification of the nominal failure probabilities assigned to failure of the turbines to run given water ingestion. In response to RAI 08.d (Reference 3), the licensee stated that the HPCI and RCIC turbines were not specifically designed to continue running while ingesting liquid. However, the licensee cited evidence that the turbines would operate given water ingestion such as: (1) a third party vendor letter about a test performed for similar turbines and (2) thermohydraulic modelling analysis done to explain the successful operation of one of the Fukushima Daiichi RCIC turbines in excess of 24 hours during the Fukushima accident. The licensee also explained that based on the judgment of operators and system managers, a nominal value of 0.05 was assumed in the PRA models as the failure probability that RCIC or HPCI turbines failed to start with water in the steam lines. The NRC staff concluded that these studies indicate that the turbines may continue to function but did not provide sufficient information to select a failure probability and requested additional information about the sensitivity of the categorization results on the failure probability in RAI 08.d.01 (Reference 9).

In response to RAI 08.d.01 (Reference 6), the licensee summarized the results of a sensitivity study, which increased the RCIC and HPCI turbine failure probability given water ingestion. The licensee noted that the importance measures are relative measures, and therefore, the increases in importance for some SSCs affected by the increase is accompanied by decreases in importance of other SSCs not affected by the increase. The licensee stated that when the failure probability is increased from 0.05 to 0.5, 56 SSCs that were LSS (based on the importance measures) became HSS, and that 73 SSCs that were HSS became LSS. The licensee evaluated the 56 SSCs that moved from LSS to HSS and determined that these SSC would be assigned HSS based on the other considerations in the categorization process. The licensee did not further evaluate changes to the 73 SCC that moved from HSS to LSS because the licensee proposed to keep the 0.05 failure probability, and therefore, these 73 would remain HSS. A 0.5 failure probability is considered a conservative assumption given that the operability of these turbines after ingesting water is expected. The licensee further proposed a license commitment in its supplement dated August 22, 2018 (Reference 6), that the failure probability will be treated as a key source of uncertainty in the 50.69 categorization process. The NRC staff finds that the licensee's sensitivity study provided in response to RAI 8.d.01, which increased the failure probability by a factor of 10, sufficiently demonstrates that the uncertainty associated with the RCIC and HPCI turbine failure probability would have a minimal impact on the 50.69 categorization even if the failure probability increased from 0.05 to 0.5, and therefore, is acceptable for the application.

The licensee stated that when the residual heat removal (RHR) system cross-tie is implemented for accidents in which net-positive suction head (NPSH) is lost, a "nominal failure probability" is applied, providing "a slight conservative bias." In RAI 08.e, the NRC staff requested explanation of the RHR cross-tie and how it is modeled in the PRA. In response to RAI 08.e.i (Reference 3), the licensee stated that as a result of the extended power uprate modifications, containment accident pressure is no longer required to ensure adequate NPSH for pumps taking suction from the torus. The licensee stated, however, that the success of this option is dependent on using the RHR cross-tie within one hour of a large-break loss-of-cooling accident coincident with a containment failure. The licensee described the model update made during the extended power uprate to model the RHR cross-tie and states that the update does not constitute a PRA upgrade. In response to RAI 08.e.ii, the licensee further explained that NPSH can be maintained through operator action to utilize RHR cross-tie and to throttle injection flow, which is a proceduralized operator action. The licensee stated that a human error probability (HEP) of 0.1 was assigned to this event to reflect the uncertainty about operator success in this scenario. The NRC staff notes that a HEP value of 0.1 can be used as a HEP screening value. In addition, in accordance with NEI 00-04, HEP sensitivity studies are to be performed during the categorization. Because the assigned failure probability of the cited action of 0.1 appears to be a reasonable reflection of uncertainty about success of the action, and because the HEP sensitivity studies specified in NEI 00-04, Tables 5-2 and 5-3, are expected to address any HEP uncertainties, the NRC staff concludes that this uncertainty is adequately addressed for this application.

The licensee stated that nominal failure probabilities are assigned to the low intake pond level to reflect the overall likelihood that events proceed to an unrecoverable event. In RAI 08.f, the NRC staff requested a description and justification for how the cited nominal failure probabilities were derived, as well as a description of actions that operators are required to take when the intake pond level is low, and confirmation that these are proceduralized steps. In response to RAI 08.f (Reference 3), the licensee described an approach for determining the likelihood of a "loss of intake event" caused by low intake pond level. The licensee presented and described an event tree initiated by an intake suction degradation event, which is considered a precursor to a low intake event. The licensee explained that such precursor events have occurred three times in the history of the site. The licensee explained that operator responses to low river/pond events are specified in plant procedures. Because the licensee modelled the loss of intake event caused by low intake pond level using an event tree with conservatively or reasonably assigned branch point probabilities, the NRC staff concludes that this uncertainty is adequately addressed for this application.

The licensee identifies dependent HEP values as a source of uncertainty but explained that NEI 00-04 requires sensitivity studies on HEPs as part of the 10 CFR 50.69 categorization process. In response to RAI 08.g (Reference 3), the licensee further explained that dependent HEP will be included as part of the HEP sensitivity studies that will be performed as part of the 10 CFR 50.69 categorization process, consistent with the guidance in NEI 00-04; therefore, the NRC staff concludes that this uncertainty is adequately addressed for this application.

The licensee stated that its approach to modeling water hammer events in the RHR system when in suppression pool cooling mode was reasonable, but it did not describe or justify the approach. In RAI 08.h, the NRC staff requested a description of, and basis for, the approach or to show that the modeling has no impact on the application. In response to RAI 08.h

(Reference 3), the licensee explained that the approach was developed based on a 1983 water hammer event and industry data assessment that involved assigning split fractions to various leakage levels. The licensee explained that it also performed a sensitivity study in which the probabilities for the damage outcomes were increased by a factor of five. The results of the sensitivity study showed that no new PRA basic events exceeded the Fussell-Vesely or Risk Achievement Worth importance value thresholds for high safety-significance in 10 CFR 50.69. Because the licensee's treatment of the uncertainty is based on industry data, and because the licensee performed a sensitivity study that shows 10 CFR 50.69 categorization is not sensitive to this uncertainty, the NRC staff concludes that this uncertainty is adequately addressed for this application.

The licensee reported its evaluations and resolutions of assumptions and key uncertainties, providing implementation items #9, #10, and #11 to adjust the DG cooling fan success criteria to update the pipe rupture frequencies in the internal flooding PRA model and to remove credit for core melt arrest in-vessel at high RPV pressure, and provided a commitment to perform sensitivity studies and to continue to evaluate new data about the failure probability of the HPCI/RCIC turbine pumps after ingesting water. Therefore, the NRC staff finds that the licensee searched for, identified, and resolved sources of uncertainty in its internal and fire PRAs, consistent with the relevant guidance in NUREG-1855 (Reference 32) and EPRI document TR-1016737 (Reference 33).

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-basis functions and functions credited for mitigation and prevention of severe accidents.

As described in the LAR and further clarified in the supplement dated October 26, 2017 (Reference 2), 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); and
- 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. 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 36), during the IPEEE (Reference 37). The SMA is a screening method that does not quantify CDF. Instead, 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.

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)

The licensee evaluated external hazards initially 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 licensee's IPEEE external hazard analysis used a progressive screening approach and concluded that all 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 ASME/ANS RA-Sa-2009 (Reference 14) and screened all external hazards except seismic events.

In response to RAI 10 (Reference 3), requesting 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 NEI 00-04 Figure 5-6. The licensee stated that this process had been completed for external flooding in an earlier flood hazard reevaluation report, and that the NRC staff issued an updated version of its assessment of this evaluation in a letter dated November 6, 2017 (Reference 38), indicating that no SSCs are credited in the screening of the external flooding hazard. In addition, the licensee stated that either available physical margin exists, or, where water ingress is expected, all external flooding mechanism resulted in water surface elevations below the design-basis protection of the plant. However, the evaluation letter refers to "flood protection features," which are credited to reliably maintain key safety functions. Therefore, in RAI 10.b.01 (Reference 8), the NRC staff requested clarification of the "flood protection features" referred to in the staff assessment constitute SSCs credited for screening the external flooding hazard. In response to RAI 10.b.01 (Reference 5), the licensee clarified that screening of local intense precipitation floods required credit for permanently installed doors to slow the ingress of water and that if the credited doors are categorized, then they will be categorized as HSS in accordance with the guidance in NEI 00-04 Figure 5-6. In addition, the licensee stated that the IDP will be informed of the basis for the local intense precipitation screening during the categorization process. The NRC staff concludes the licensee's treatment of external flood screening is acceptable because the SSCs that participate in the screening of the external flooding would be categorized HSS in accordance with NRC-approved guidance.

In RAI 10.c, the NRC staff requested discussion of SSCs, if any, credited for screening extreme winds and tornadoes, and whether the guidance in NEI 00-04 Figure 5-6 will be applied to the extreme wind and tornado hazard. The NRC staff also requested explanation of the impact on the application of the licensee's current effort to assess tornado missile protection in response to Regulatory Issue Summary (RIS) 2015-06, "Tornado Missile Protection" (Reference 39). In response to RAI 10.c (Reference 4), the licensee explained that there were no SSCs credited in the screening of extreme winds and tornadoes, including passive or active components, other than Category 1 structures. The licensee explained that, according to the process described in Section 5.4 and Figure 5-6 of NEI 00-04, all SSCs were credited for screening of extreme wind and tornados will be categorized HSS, and the basis for that conclusion will be identified. The licensee explained that in order to ensure current efforts to respond to RIS 2015-06 are reflected in the 10 CFR 50.69 categorization process, the licensee proposed updated implementation item #12 to complete necessary actions (e.g., analyses, modifications, etc.) to screen tornado missile hazards in accordance with the original LAR dated August 30, 2017, prior to the adoption of 10 CFR 50.69. Page 58 of the LAR Attachment 4 states that if additional tornado missile protection vulnerabilities are discovered as part assessing tornado missile protection in response to RIS 2015-06, then this information will be used to update the screening process.

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

Paragraph 50.69(c)(1)(ii) of 10 CFR requires a licensee 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. Consistent with NEI 00-04, the licensee proposes to use the shutdown safety assessment process based on NUMARC 91-06 (Reference 11). The guidance in 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. The guidance in 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 11 (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. The guidance in 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 failure of the SSC 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 addresses the passive function of active components, such as the pressure/liquid retention function of the body of a motor-operated valve.

In Section 3.1.2 of 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 31). 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 concrete containment and metallic containment items), using a modification of ASME Code Case N-660, "Risk-Informed Safety Classification for Use in Risk-Informed Repair/Replacement Activities, Section XI, Division 1." 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. In RAI 15 (Reference 8), 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 15 (Reference 3), the licensee stated that it will apply the process for the passive categorization of Class 2, Class 3, and non-Code class components. The licensee proposed implementation item #13, 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, which will result in HSS for its risk-informed safety classification and 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 the 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. 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 Plan to assess shutdown risk
- ANO-2 (Reference 31) passive categorization method to assess passive component risk for Class 2 and Class 3 SSCs and their associated supports

The licensee proposed the addition of a license condition for the implementation of 10 CFR 50.69. The license condition (Reference 4) identifies thirteen implementation items that shall be completed prior to the implementation of the 10 CFR 50.69 categorization process, ten of which are updates to the fire PRA, one regarding additional sensitivity studies that will be completed as part of the categorization process, and two of which are associated with categorization that are not specifically defined in NEI 00-04:

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 03.a.

ii. Update the transient floor ratios (FARs) in the fire PRA to consider the treatment of obstructed floor space to provide a more accurate distribution of transient fire ignition frequency, as indicated in response to RAI 03.b.

iii. Update risk significant fire PRA scenarios for ignition sources capable of being modelled with a two-point modeling approach, as indicated in response to RAI 03.c and RAI 03.c.01.

iv. Review potentially vulnerable fire wrap configurations to identify which are subject to mechanical damage and update the fire PRA to ensure that fire wrap configurations are not credited in fire scenarios that could subject it to mechanical damage, as indicated in response to RAI 03.d.

v. Perform another fire PRA sensitivity study as part of the categorization process that assumes credit for immediate manual suppression, as indicated in response to RAI 04.a. The previous fire PRA sensitivity studies from NEI 00-04 Table 5-3 included the sensitivity study to remove all credit for manual suppression.

vi. Remove the sources of uncertainty associated with implementation of FLASHCAT in the fire PRA introduced (through use of generic parameters from NUREG-7010 Vol. 1 and weighted averages of parameters for cables located within the physical analysis units in which the scenarios implementing the FLASHCAT model were located) and instead base the values for these parameters (mass per unit length and plastic mass fraction) on the scenario specific set of cables that are located within the cable trays analyzed using the FLASHCAT model, as indicated in response to RAI 04.a and RAI 04.d.

vii. Apply the NUREG-1921 methodology in identifying undesired operator actions and use it to incorporate any identified actions into the fire PRA and perform a focused-scope peer review of the application of this methodology including a resolution of any new F&Os resulting from the focused scope review, as indicated in response to RAI 05.a.

viii. Update the fire PRA model to address breaker coordination in non-safety related power supplies credited in the model by assuming failure of the power supply, including accounting for opening of the power supply upstream breaker that may occur due to the potential lack of coordination between it and the downstream breaker associated with the damaged power cable, when the power cable within the circuits of concern are identified to be damaged by fire scenarios, or perform additional analysis to determine that circuits are coordinated, as indicated in response to RAI 05.b and RAI 05.b.01.

ix. Update the PRA model to account for the requirement for two EDG [emergency DG] cooling fans during periods when the outdoor temperature at Peach Bottom are above the design temperature of 80°F, as indicated in response to RAI 08.a.

x. Update the pipe rupture frequencies in the internal flooding PRA to the most recent EPRI pipe rupture frequencies, as indicated in response to RAI 07.a.

xi. Remove credit for core melt arrest in-vessel at high reactor pressure vessel (RPV) pressure conditions, as indicated in response to RAI 07.a.

xii. Complete any necessary actions (e.g., analyses, modifications, etc.) to screen tornado missile hazards in accordance with the original LAR dated August 30, 2017.

xiii. Designate as high safety-significant (HSS) for passive categorization all ASME Code Class 1 SSCs with a pressure retaining function, as well as supports which results in HSS for its risk-informed safety classification, and cannot be changed by the IDP, as indicated in response to RAI 15.

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.

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

NEI 00-04, Section 6.0, provides guidance on assessment of DID. NEI 00-04 Figure 6-1 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 clarifies 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. The NRC staff finds the licensee's 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 in.

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 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 preliminary HSS. In RAI 09.d (Reference 8), 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 09.d, the licensee explained that the safety-significance of functions will be preliminary HSS only if it is supported by a component determined to be HSS from a PRA-based assessment (i.e., for PBAPS, 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 their risk significance. The licensee 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 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 3.10 of this SE, assure that the potential cumulative risk increase from the categorization is maintained acceptably low. The performance monitoring process monitors the component performance to ensure that potential increases in failure rates of categorized components are detected and addressed before reaching the rate assumed in the sensitivity study. The 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 maintained acceptably low as required by 10 CFR 50.69(c)(1)(iv) .

3.9 Integrated Decision-Making 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, and 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.

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 09 (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 09. As shown 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 recategorized 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) may be categorized as LSS by the IDP.

In response to RAI 09 (Reference 3) with respect to the footnote to Table 1 (Reference 3), the licensee provided information on how the seven qualitative criteria in Section 9.2 of NEI 00-04 will be used by the licensee to determine the safety-significance of the system functions. The licensee stated that the final assessment of the seven considerations is the direct responsibility of the IDP and 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. The NRC staff finds the licensee's proposed use of the seven qualitative questions in the 10 CFR 50.69 categorization process is consistent with the guidance in NEI 00-04, and therefore, acceptable.

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, the IDP may recategorize 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, confidence that sufficient safety margins are maintained for SSCs categorized as RISC-3. Safety margins are addressed through an integrated engineering evaluation that would nominally be addressed by the IDP. As discussed in NEI 00-04, the only LSS SSC requirements that are relaxed for RISC-3 (LSS) SSCs are those related to treatment, not design or capability, and 10 CFR 50.69(d)(2)(i) requires the licensee ensures, with reasonable confidence, that RISC-3 SSCs remain capable of performing their safety-related functions under design-basis conditions. Loss of capability (i.e., loss of sufficient safety margins) that would be contrary to the rule should be avoided by the licensee's program, and, if discovered by the license or by the inspection process, would require actions to correct. Therefore, the NRC staff finds that a 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 and that the IDP need not explicitly consider safety margins, consistent with the discussion in the guidance endorsed by RG 1.201.

3.10 Program Documentation and Change Control (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 includes Section 11 on program documentation and change control and Section 12 on periodic review. These sections are described in NEI 00-04 and the LAR with respect to satisfying rule 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 10 CFR 50.69(e) for periodic updates. As provided in RG 1.200, the NRC staff's 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 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. 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 requirements 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

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 documentation 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), and therefore, the NRC staff finds them acceptable.

The NRC staff finds that the change control and performance monitoring of categorized SSCs and PRA updates will capture and evaluate component failures to identify significant changes in the failure probabilities. 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 that the process in NEI 00-04 and the LAR will meet 10 CFR 50.69(e) and 10 CFR 50.69(f) of 10 CFR, respectively, and 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).

4.0 SECTION 50.69 OF 10 CFR 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 staff has concluded that the application includes a process description that satisfies the requirements of 10 CFR 50.69(c). In the LAR and in the licensee's responses to the NRC staff's RAIs, there were certain specific actions that the NRC staff identified as being necessary to support the conclusion that the proposed program met the requirements in 10 CFR 50.69 and the guidance in RG 1.201 and NEI 00-04. The licensee did not complete some of the actions. Additional actions (e.g., final procedures and proposed alternative treatment) need not, and have not, been developed, submitted, or reviewed by the staff but will be completed before implementation of the program as specified in the 10 CFR 50.69 rule.

The NRC staff's finding on the acceptability of the PRA evaluation in the proposed process is dependent on the completion of ten changes to the PRA, the addition of a sensitivity study to the studies summarized in Tables 5-2 and 5-3 of NEI 00-04, and two items associated with categorization that are not specifically defined in NEI 00-04. These 13 items are identified as "Peach Bottom 50.69 PRA Implementation Items" in Attachment 2 of the licensee's letter dated June 6, 2018 (Reference 4). Other changes that were described by the licensee are less important and are similar to occasional future changes to the PRA and PRA methods that may occur over time, and therefore, can be addressed and resolved using the licensee's periodic review process.

In its August 10, 2018, letter (Reference 5), 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 Unit [2 or 3] License Amendment No. [321 or 324] dated October 25, 2018.
>
> Exelon will complete the implementation items listed in Attachment 2 of Exelon letter to NRC dated June 6, 2018 prior to implementation of 10 CFR 50.69. All issues identified in the attachment will be addressed and any associated changes will be made, focused-scope peer reviews will be performed on changes that are PRA upgrades as defined in the PRA Standard (ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, Revision 2), and any findings will be resolved and reflected in the PRA of record prior to implementation of the 10 CFR 50.69 categorization process.
>
> Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from a seismic margins approach to a seismic probabilistic risk assessment approach).

The 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 accepted 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. 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. An onsite audit or future inspections could be subject to appropriate NRC enforcement action, as they are part of the proposed license conditions.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the staff notified the Pennsylvania State official on September 24, 2018, of the proposed issuance of the amendments. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. 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 November 21, 2017 (82 FR 55404), 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

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2. Barstow J., Exelon Generation Company, LLC, letter to U.S. Nuclear Regulatory Commission, "Peach Bottom Atomic Power Station, Units 2 and 3, Renewed Facility Operating License Nos. DPR-44 and DPR-56, NRC Docket Nos. 50-277 and 50-278, Supplement to Application to Adopt 10 CFR 50.69 Risk-Informed Categorization and Treatment of SSScs for NPPs," October 24, 2017 (ADAMS Accession No. ML17297B521).

3 Barstow J., Exelon Generation Company, LLC, letter to U.S. Nuclear Regulatory Commission, "Peach Bottom Atomic Power Station, Units 2 and 3, Renewed Facility Operating License Nos. DPR-44 and DPR-56, NRC Docket Nos. 50-277 and 50-278, Response to RAI for Application to Adopt 10 CFR 50.69 Risk-informed categorization and treatment of SSCs," May 7, 2018 (ADAMS Accession No. ML18128A009).

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9 Tobin, Jennifer, U.S. Nuclear Regulatory Commission, E-Mail to Helker, David, P., Exelon Generation Company, LLC, "Peach Bottom Units 2 and 3 - Request for Additional Information - Adopt 50.69 License Amendment (EPID L-2017-LLA-0281)," July 10, 2018, (ADAMS Accession No. ML18192A119).

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

12 U.S. Nuclear Regulatory Commission, "Guidelines For Categorizing Structures, Systems, And Components In Nuclear Power Plants According To Their Safety Significance, For Trial Use," Regulatory Guide 1.201, Revision 1, May 2006 (ADAMS Accession No. ML061090627).

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