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

June 26, 2020

Ms. Cheryl A. Gayheart
Regulatory Affairs Director
Southern Nuclear Operating Co., Inc.
3535 Colonnade Parkway
Birmingham, AL 35243

SUBJECT: EDWIN I. HATCH NUCLEAR PLANT, UNIT NOS. 1 AND 2, ISSUANCE OF AMENDMENTS NOS. 305 AND 250, REGARDING ADOPTION OF 10 CFR 50.69, "RISK-INFORMED CATEGORIZATION AND TREATMENT OF STRUCTURES, SYSTEMS AND COMPONENTS FOR NUCLEAR POWER REACTORS" (EPID L-2018-LLA-0175)

Dear Ms. Gayheart:

The Nuclear Regulatory Commission (NRC, the Commission) has issued the enclosed Amendment No. 305 to Renewed Facility Operating License (RFOL) No. DPR-57 and Amendment No. 250 to RFOL No. NPF-5 for the Edwin I. Hatch Nuclear Plant, Unit Nos 1 and 2, respectively. The amendments consist of changes to the RFOLs in response to your application dated June 7, 2018, as supplemented by letters dated July 16 and December 18, 2019, and June 2, 2020.

The amendments revise the RFOLs to allow for the implementation of the provisions of Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.69, "Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors."

The provisions of 10 CFR 50.69 allow adjustment of the scope of structures, systems and components (SSCs) subject to special treatment requirements (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation) based on an integrated and systematic risk-informed method of categorizing SSCs according to their safety significance. For SSCs determined to be of low safety significance, alternative treatment requirements can be implemented in accordance with this regulation. For equipment determined to be of high safety significance, requirements may not be changed.

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,

/RA/

John G. Lamb, Senior Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-321 and 50-366

Enclosures:
1. Amendment No. 305 to DPR-57
2. Amendment No. 250 to NPF-5
3. Safety Evaluation

cc: Listserv

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

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

DOCKET NO. 50-321

EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE DPR-57

Amendment No. 305
Renewed License No. DPR-57

1. The Nuclear Regulatory Commission (NRC, the Commission) has found that:

   A. The application for amendment to the Edwin I. Hatch Nuclear Plant, Unit No. 1 (the facility), Renewed Facility Operating License No. DPR-57 filed by Southern Nuclear Operating Company, Inc. (the licensee), acting for itself, Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia (the owners), dated June 7, 2018, as supplemented by letters dated July 16 and December 18, 2019, and June 2, 2020, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as 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 set forth in 10 CFR Chapter I;

   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 Renewed Facility Operating License No. DPR-57 is hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraphs 2.C.(2) and 2.C.(11) of Renewed Facility Operating License No. DPR-57 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications (Appendix A) and the Environmental Protection Plan (Appendix B), as revised through Amendment No. 305, are hereby incorporated in the renewed license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(11) Risk-Informed Categorization and Treatment of Structures, System and Components

1) Southern Nuclear Operating Company 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 SSCs specified in Renewed License Amendment No. 305, dated June 26, 2020.

2) 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) Prior to implementation of the Renewed License Amendment No. 305, dated June 26, 2020, Southern Nuclear Operating Company shall update the Probabilistic Risk Assessment (PRA) models to reflect the as-built, as-operated, and as-maintained plant and shall ensure the risk acceptance guidelines found in Regulatory Guide (RG) 1.174, Revision 3 are met.

3. This license amendment is effective as of its date of issuance and shall be implemented within 90 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley

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Date: 2020.06.26 09:57:06 -04'00'

Michael T. Markley, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed Facility
Operating License No. DPR-57
and Technical Specifications

Date of Issuance: June 26, 2020

ATTACHMENT TO LICENSE AMENDMENT NO. 305

EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 1

RENEWED FACILITY OPERATING LICENSE NO. DPR-57

DOCKET NO. 50-321

Replace the following pages of the License and the Appendix A Technical Specifications (TSs) with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove Pages	Insert Pages
License	License
4	4
11	11
TSs	TSs
N/A	N/A

for sample analysis or instrument calibration, or associated with radioactive apparatus or components.

(6) Southern Nuclear, pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

A. This renewed license shall be deemed to contain, and is subject to, the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Section 50.54 of Part 50, and Section 70.32 of Part 70; all applicable provisions of the Act and the rules, regulations, and orders of the Commission now or hereafter in effect; and the additional conditions specified or incorporated below:

(1) <u>Maximum Power Level</u>

Southern Nuclear is authorized to operate the facility at steady-state reactor core power levels not in excess of 2,804 megawatts thermal.

(2) <u>Technical Specifications</u>

The Technical Specifications (Appendix A) and the Environmental Protection Plan (Appendix B), as revised through Amendment No. 305, are hereby incorporated in the renewed license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

The Surveillance Requirement (SR) contained in the Technical Specifications and listed below, is not required to be performed immediately upon implementation of Amendment No. 195. The SR listed below shall be successfully demonstrated before the time and condition specified:

SR 3.8.1.18 shall be successfully demonstrated at its next regularly scheduled performance.

(3) <u>Fire Protection</u>

Southern Nuclear Operating Company shall implement and maintain in effect all provisions of the approved fire protection program that comply with 10 CFR 50.48(a) and 10 CFR 50.48(c), as specified in the licensee amendment request dated April 4, 2018, supplemented by letters dated May 28, August 9, October 7, and December 13, 2019, and February 5, and March 13, 2020, and as approved in the NRC safety evaluation (SE) dated June 11, 2020. Except where NRC approval for changes or deviations is required by 10 CFR 50.48(c), and provided no other regulation, technical specification, license condition or requirement would require prior NRC approval, the licensee may make changes to the fire protection program without prior approval of the Commission if those changes satisfy the provisions set forth in 10 CFR 50.48(a) and 10 CFR 50.48(c), the change does not require a change to a technical specification or a license condition, and the criteria listed below are satisfied.

c. The first performance of the periodic measurement of CRE pressure, Specification 5.5.14.d, shall be within 24 months, plus the 6 months allowed by SR 3.0.2, from the date of the most recent successful pressure measurement test.

(11) <u>10 CFR 50.69 Risk-Informed Categorization</u>

Southern Nuclear Operating Company 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) specified in the Renewed License Amendment No. 305, dated June 26, 2020.

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).

Prior to implementation of the Renewed License Amendment No. 305, dated June 26, 2020, Southern Nuclear Operating Company shall update the Probabilistic Risk Assessment (PRA) models to reflect the as-built, as-operated, and as-maintained plant and shall ensure the risk acceptance guidelines found in Regulatory Guide (RG) 1.174, Revision 3 are met

D. Southern Nuclear shall not market or broker power or energy from Edwin I. Hatch Nuclear Plant, Unit 1.

3. This renewed license is effective as of the date of issuance and shall expire at midnight, August 6, 2034.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Samuel J. Collins, Director
Office of Nuclear Reactor Regulation

Attachments:
Appendix A – Technical Specifications
Appendix B – Environmental Protection Plan

Date of Issuance: January 15, 2002

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

DOCKET NO. 50-366

EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE NPF-5

Amendment No. 250
Renewed License No. NPF-5

1. The Nuclear Regulatory Commission (NRC, the Commission) has found that:

    A. The application for amendment to the Edwin I. Hatch Nuclear Plant, Unit No. 2 (the facility), Renewed Facility Operating License No. NPF-5 filed by Southern Nuclear Operating Company, Inc. (the licensee), acting for itself; Georgia Power Company; Oglethorpe Power Corporation; Municipal Electric Authority of Georgia; and City of Dalton, Georgia (the owners), dated June 7, 2018, as supplemented by letters dated July 16 and December 18, 2019, and June 2, 2020, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as 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 set forth in 10 CFR Chapter I;

    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 Renewed Facility Operating License No. NPF-5 is hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraphs 2.C.(2) and 2.C.(11) of Renewed Facility Operating License No. NPF-5 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications (Appendix A) and the Environmental Protection Plan (Appendix B), as revised through Amendment No. 250 are hereby incorporated in the renewed license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(11) Risk-Informed Categorization and Treatment of Structures, Systems, and Components

1) Southern Nuclear Operating Company 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 SSCs specified in Renewed License Amendment No. 250, dated June 26, 2020.

2) 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) Prior to implementation of the Renewed License Amendment No. 250, dated June 26, 2020, Southern Nuclear Operating Company shall update the Probabilistic Risk Assessment (PRA) models to reflect the as-built, as-operated, and as-maintained plant and shall ensure the risk acceptance guidelines found in Regulatory Guide (RG) 1.174, Revision 3 are met.

3. This license amendment is effective as of its date of issuance and shall be implemented within 90 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley

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Date: 2020.06.26 09:57:59 -04'00'

Michael T. Markley, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed Facility
Operating License No. NPF-5
and Technical Specifications

Date of Issuance: June 26, 2020

ATTACHMENT TO LICENSE AMENDMENT NO. 250

EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 2

RENEWED FACILITY OPERATING LICENSE NO. NPF-5

DOCKET NO. 50-366

Replace the following pages of the License and the Appendix A Technical Specifications (TSs) with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove Pages	Insert Pages
License	License
4	4
11	11
TSs	TSs
N/A	N/A

(6) Southern Nuclear, pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This renewed license shall be deemed to contain, and is subject to, the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Section 50.54 of Part 50, and Section 70.32 of Part 70; all applicable provisions of the Act and the rules, regulations, and orders of the Commission now or hereafter in effect; and the additional condition[2] specified or incorporated below:

(1) Maximum Power Level

Southern Nuclear is authorized to operate the facility at steady-state reactor core power levels not in excess of 2,804 megawatts thermal, in accordance with the conditions specified herein.

(2) Technical Specifications

The Technical Specifications (Appendix A) and the Environmental Protection Plan (Appendix B), as revised through Amendment No. 250, are hereby incorporated in the renewed license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Additional Conditions

The matters specified in the following conditions shall be completed to the satisfaction of the Commission within the stated time periods following the issuance of the renewed license or within the operational restrictions indicated. The removal of these conditions shall be made by an amendment to the license supported by a favorable evaluation by the Commission.

(a) Fire Protection

Southern Nuclear Operating Company shall implement and maintain in effect all provisions of the approved fire protection program that comply with 10 CFR 50.48(a) and 10 CFR 50.48(c), as specified in the licensee amendment request dated April 4, 2018, supplemented by letters dated May 28, August 9, October 7, and December 13, 2019, and February 5, and March 13, 2020, and as approved in the NRC safety evaluation (SE) dated June 11, 2020. Except where NRC approval for changes or deviations is required by 10 CFR 50.48(c), and provided no other regulation, technical specification, license condition or requirement would

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2. The original licensee authorized to possess, use, and operate the facility was Georgia Power Company (GPC). Consequently, certain historical references to GPC remain in certain license conditions.

(h) TSTF-448 Control Room Habitability

Upon implementation of the Amendments adopting TSTF-448, Revision 3, the determination of control room envelope (CRE) unfiltered air inleakage as required by SR 3.7.4.4, in accordance with TS 5.5.14.c.(i), the assessment of CRE habitability as required by Specification 5.5.14.c.(ii), and the measurement of CRE pressure as required by Specification 5.5.14.d, shall be considered met. following implementation:

i) The first performance of SR 3.7.4.4, in accordance with Specification 5.5.14.c.(i), shall be within the next 18 months.

ii) The first performance of the periodic assessment of CRE habitability, Specification 5.5.14.c.(ii), shall be within 3 years, plus the 9-month allowance of SR 3.0.2, of the next successful tracer gas test.

iii) The first performance of the periodic measurement of CRE pressure, Specification 5.5.14.d, shall be within 24 months, plus the 6 months allowed by SR 3.0.2, from the date of the most recent successful pressure measurement test.

(i) 10 CFR 50.69 Risk-Informed Categorization

Southern Nuclear Operating Company 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) specified in the Renewed License Amendment No. 250 dated June 26, 2020.

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).

Prior to implementation of the Renewed License Amendment No. 250 dated June 26, 2020, Southern Nuclear Operating Company shall update the Probabilistic Risk Assessment (PRA) models to reflect the as-built, as-operated, and as-maintained plant and shall ensure the risk acceptance guidelines found in Regulatory Guide (RG) 1.174, Revision 3 are met.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 305 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-57

AND

AMENDMENT NO. 250 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-5

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

EDWIN I. HATCH NUCLEAR PLANT, UNIT NOS. 1 AND 2

DOCKET NOS. 50-321 AND 50-366

1.0 INTRODUCTION

By letter dated June 7, 2018 (Reference 1), as supplemented by letter dated July 16 (Reference 2) and December 18, 2019 (Reference 3), and June 2, 2020, (Reference 4) Southern Nuclear Company (SNC, the licensee) submitted a license amendment request (LAR) for the Edwin I. Hatch Nuclear Plant (Hatch), Unit Nos. 1 and 2. The licensee proposed to add a new license condition to the Hatch Renewed Facility Operating Licenses (RFOLs) to allow the implementation of Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.69, "Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors." The provisions of 10 CFR 50.69 allow adjustment of the scope of structures, systems, and components (SSCs) subject to special treatment requirements (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation) based on an integrated and systemic risk-informed process for categorizing SSCs according to their safety significance. For equipment determined to be of low safety significance, alternative treatment requirements can be implemented in accordance with this regulation. For equipment determined to be of high safety significance, requirements may not be changed or will be enhanced.

The proposed Hatch RFOLs license condition that would allow for the implementation of 10 CFR 50.69 would state:

1) Southern Nuclear Operating Company 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 SSCs specified in Renewed License Amendment No. 305 for Hatch, Unit 1, and Amendment No. 250 for Hatch, Unit 2, dated June 26, 2020.

2) 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) Prior to implementation, Renewed License Amendment No. 305 for Hatch Unit 1, and Amendment No. 250 for Hatch, Unit 2, dated June 26, 2020, Southern Nuclear Operating Company shall update the Probabilistic Risk Assessment (PRA) models to reflect the as-built, as-operated, and as-maintained plant and shall ensure the risk acceptance guidelines found in Regulatory Guide (RG) 1.174, Revision 3 are met.

On March 25-26, 2019, the U.S. Nuclear Regulatory Commission (NRC) staff and its contractors from the Pacific Northwest National Laboratory (PNNL) participated in a remote regulatory audit. The NRC staff performed the audit to ascertain the information needed to support its review of the application and develop requests for additional information (RAIs), as needed (Reference 5). By email dated April 17, 2019 (Reference 6) the NRC staff ("the staff") requested additional information from the licensee. By letters dated July 16 and December 18, 2019, the licensee submitted supplements responding to the request for additional information.

The supplements dated July 16 and December 18, 2019, and June 2, 2020, 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 March 26, 2019 (84 FR 11341).

2.0 REGULATORY EVALUATION

2.1 Applicable Regulations

The NRC staff reviewed the LAR, as supplemented 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 NRC staff considered the following principles and regulatory requirements during its review of the proposed changes.

The NRC issued 10 CFR 50.69, in the *Federal Register* on November 22, 2004 (69 FR 68008), which became effective on December 22, 2004. The risk-informed approach discussed in the rule establishes an alternative scope of SSCs subject to special treatment requirements that uses both risk and traditional deterministic methods in a blended approach. Section 50.69(b)(2)(iv) requires licensees to consider potential effects of common-cause interaction susceptibility and potential impacts from known degradation mechanisms. The rule focuses on common-cause effects because significant increases in common-cause failures (CCFs) could invalidate the evaluation performed to show that any potential change in risk from implementation of 10 CFR 50.69 would be small.

Use of a probabilistic approach allows consideration of a broader set of potential challenges to safety, thus 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. PRAs address credible initiating events by assessing the event frequency. Mitigating system reliability is then assessed, including the potential for CCFs.

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 perform their design-basis functions. For SSCs categorized as low safety-significant (LSS), alternative treatment requirements may be implemented in accordance with the regulation. For SSCs determined to be of high safety-significant (HSS), requirements may not be changed.

Section 50.69 of 10 CFR contains requirements regarding how a licensee categorizes SSCs using a risk-informed process; adjusts treatment requirements consistent with the relative significance of the SSC; and manages the process over the lifetime of the plant. A risk-informed categorization process is employed to determine the safety significance of SSCs and place the SSCs into one of four RISC categories. The determination of safety significance is performed by an integrated decision-making process, which uses both risk insights and traditional engineering insights. The safety functions include the design-basis functions (derived from the "safety-related" definition, which includes external events), 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 category into which the SSC is categorized. The rule requires licensees to adjust the categorization or treatment processes, as appropriate, in response to SSC performance information obtained as part of the treatment process.

Section 50.69 of 10 CFR does not allow for the elimination of SSC functional requirements or allow equipment that is required by the deterministic design basis to be removed from the facility. Instead, the rule enables licensees and the NRC staff to focus resources on SSCs that make a significant contribution to plant safety. This was stated explicitly in the Statement of Consideration for the final rule.[1]

For SSCs that are categorized as HSS, existing treatment requirements are maintained or potentially enhanced. Conversely, for SSCs categorized as LSS that do not significantly contribute to plant safety on an individual basis, the regulation allows an alternative risk-informed approach to treatment that provides a reasonable level of confidence that these SSCs will satisfy functional requirements. Implementation of 10 CFR 50.69 allows licensees to improve focus on equipment that is categorized as HSS.

1 Final Rule, Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors, 69 FR 68008, 68011 (Nov. 22, 2004).

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

RISC-1: Safety-related SSCs that perform safety-significant functions[2]

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

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

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

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

Section 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 [DID].

(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

2 Nuclear Energy Institute (NEI) 00-04, Revision 0, "10 CFR 50.69 SSC Categorization Guideline," July 2005 (Reference 8), uses the term "high-safety-significant" 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.

changes in treatment permitted by implementation of 10 CFR 50.69(b)(1) and (d)(2) are small.

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

Section 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.

Section 50.69(b)(3) of 10 CFR states that the Commission will approve a licensee's implementation of this section if it determines that the process for categorization satisfies the requirements of 10 CFR 50.69(c) by issuing a licensee amendment approving the licensee's use of this section. Licensees may voluntarily comply with 10 CFR 50.69 as an alternative to compliance with the following requirements of 10 CFR 50.69(b)(1) for LSS SSCs:

(i) CFR Part 21,
(ii) The 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) The inservice testing requirements of 10 CFR 50.55a(f); the inservice inspection, and repair and replacement (with the exception of fracture toughness) requirements for ASME Class 2 and 3 SSCs in 10 CFR 50.55a(g); and the electrical component quality and qualification requirements in Section 4.3 and 4.4 of IEEE 279, and Sections 5.3 and 5.4 of IEEE 603-1991, as incorporated by reference in 10 CFR 50.55a(h),
(vi) 10 CFR 50.65, except for section (a)(4),
(vii) 10 CFR 50.72,
(viii) 10 CFR 50.73,
(ix) Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50.
(x) The Type B and Type C leakage testing requirements in both Option A and B of Appendix J 10 CFR Part 50, for penetrations and valves meeting specified criteria, and
(xi) Appendix A to 10 CFR Part 100 Sections VI(a)(1) and VI(a)(2), to the extent these regulations require qualification testing and specific engineering methods to demonstrate that SSCs are designed to withstand the Safe Shutdown Earthquake and Operating Basis Earthquake.

Finally, 10 CFR 50.69(b)(2) specifies the content of the license amendment request to implement 10 CFR 50.69, which includes a description of the process for categorization, and the results of the PRA review process conducted to meet 10 CFR 50.69(c)(1)(i).

2.2 Regulatory Guidance and Staff Review Plans

The NRC staff considered the following guidance during its review of the proposed changes.

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 7), describes a method that the NRC considers acceptable for complying with the requirements of 10 CFR 50.69. RG 1.201 endorses Nuclear Energy Institute (NEI) 00-04, Revision 0 (Reference 8), as an acceptable approach for use in categorizing SSCs with associated clarifications, and limitations and conditions. RG 1.201, Revision 1, states that the applicant is expected to document, at a minimum, the technical adequacy of the internal initiating events PRA. Licensees may use either PRAs or alternative approaches for hazards other than internal initiating events. RG 1.201, Revision 1, 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 acceptable.

RG 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities" (Reference 9) describes an acceptable approach for determining whether the acceptability of the base PRA, in total or the parts that are used to support an application, is sufficient to provide confidence in the results, such that the PRA can be used in regulatory decision making for light-water reactors. RG 1.200 endorses, with clarifications, the American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) PRA Standard ASME/ANS RA-Sa-2009 (Reference 10). Revision 2 of RG 1.200 provides guidance for determining the acceptability of a PRA by comparing the PRA to the relevant parts of ASME/ANS RA-Sa-2009 using a peer review process.

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 11), describes an approach acceptable to the NRC for developing risk-informed applications for a licensing basis change that considers engineering issues and applies risk insights. Revision 3 of RG 1.174 provides risk acceptance guidelines for evaluating the results of such evaluations.

NUREG-1855, Revision 1, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking," (Reference 12) provides guidance on how to treat uncertainties associated with PRA in risk-informed decisionmaking. The guidance provides an understanding of the uncertainties associated with PRA, their impact on the results of the PRA, and provides a pragmatic approach to addressing these uncertainties in the context of the decisionmaking.

NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Chapter 19, Section 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance" (Reference 13), provides general guidance for evaluating the technical basis for proposed risk-informed changes. Section 19.2 provides review guidance to the NRC staff that supports licensing actions submitted using industry guidance in RG 1.174 (Reference 11), and states that a risk-informed application should be evaluated to ensure the proposed changes in meet the key principles of risk-informed decision-making. Guidance to the NRC staff on evaluating PRA technical adequacy is provided in SRP, Chapter 19, Section 19.1, Revision 3, "Determining the Technical Adequacy of Probabilistic Risk Assessment for Risk-Informed License Amendment Requests After Initial Fuel Load" (Reference 14).

NRC Endorsed Guidance

The NEI 00-04, Revision 0, "10 CFR 50.69 SSC Categorization Guideline" (Reference 8), 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

decision-making 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 for the use of non-PRA approaches when PRA models have not been developed. The guidance in NEI 00-04 identifies non-PRA methods to be used as an approach, such as fire-induced vulnerability evaluation (FIVE) to address internal fire risk, seismic margin analysis (SMA) to address seismic risk, and guidance in the Nuclear Management and Resource Council (NUMARC) report NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management" (Reference 15), to address shutdown operations. As stated in RG 1.201, Revision 1, "Guidelines for Categorizing SSC in Nuclear Power Plants According to their Safety Significance", such non-PRA-type evaluations will result in more conservative categorization, in that special treatment requirements will not be allowed to be relaxed for SSCs that are relied upon in such evaluations. The degree of relief that the NRC will accept under 10 CFR 50.69 (i.e., SSCs subject to relaxation of special treatment requirements) will be commensurate with the assurance provided by the evaluations performed to assess and characterize the SSC's risk.

Sections 2 through 10 of NEI 00-04, Revision 0 describe the steps/elements of the SSC categorization process to be performed for meeting the requirements of 10 CFR 50.69, as follows:

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

Further, NEI 00-04 provides guidance to meet 10 CFR 50.69(b) with regards to the content of the license amendment request:

- Section 2 provides expectations to address 10 CFR 50.69(b)(2)(i).
- Sections 3.2 and 3.3 provide expectations and general guidance for assessment of PRA scope and technical capability to address 10 CFR 50.69(b)(2)(ii) and (iii).
- Section 8 provides specific guidance corresponding to 10 CFR 50.69(b)(iv).

Additionally, Section 11 of NEI 00-04, Revision 0, provides guidance on program documentation and change control related to the requirements of 10 CFR 50.69(f). Section 12 of NEI 00-04 provides guidance on periodic review related to the requirements in 10 CFR 50.69(e). Maintaining change control and periodic review provides confidence that all aspects of the program reasonably reflect the current plant configuration and operating practices, and applicable plant and industry operational experience, as required by 10 CFR 50.69(c)(1)(ii).

3.0 TECHNICAL EVALUATION

3.1 NRC Staff's Method of Review

The NRC staff evaluated the LAR, as supplemented, to determine if the proposed changes are consistent with the regulations and guidance discussed in Section 2 of this safety evaluation.

RG 1.174, Revision 3 (Reference 11), lays out an acceptable approach for making risk-informed decisions. Using that approach, licensing basis (LB) changes are expected to meet five key principles, laid out in Section C of RG 1.174, Revision 3. These key principles are:

Principle 1: The proposed LB change meets the current regulations unless it is explicitly related to a requested exemption.

Principle 2: The proposed LB change is consistent with the defense-in-depth philosophy.

Principle 3: The proposed LB change maintains sufficient safety margins.

Principle 4: When the proposed LB change results in an increase in risk, the increases should be small and consistent with the intent of the Commission's policy statement on safety goals for the operations of nuclear power plants.

Principle 5: The impact of the proposed LB change should be monitored by using performance measurement strategies.

The NRC staff's evaluations approach and acceptance guidelines follow from these principles.

3.2 Evaluation

The traditional engineering evaluation below addresses the first three key principles of the NRC staff's standards for risk-informed decision making and are pertinent to: (1) compliance with current regulations, (2) evaluation of defense-in-depth, and (3) evaluation of safety margins.

3.2.1 Key Principle 1: LB Change Meets the Current Regulations

The first key principle from RG 1.174 is that the proposed licensing basis change will meet the current regulations.

To ensure the five key principles of risk-informed decisionmaking were met for the adoption of 10 CFR 50.69, the NRC staff's review, as documented in this safety evaluation (SE), as used the guidance provided in NEI 00-04, Revision 0, (Reference 8), as endorsed in in RG 1.201, Revision 1 (Reference 7). The NRC staff reviewed the licensee's SSC categorization process against the process described in NEI 00-04, Revision 0, as endorsed in RG 1.201, Revision 1.

The regulation 10 CFR 50.69(b)(2)(i) states that a licensee voluntarily choosing to implement 10 CFR 50.69 shall submit an application for license amendment under 10 CFR 50.90 that contains a description of the process for categorization of RISC-1, RISC-2, RISC-3, and RISC-4 SSCs.

Section 2 of NEI 00-04, Revision 0, in part, states that the categorization process includes eight primary steps:

1. Assembly of Plant-Specific Inputs (Section 3 of NEI 00-04)
2. System Engineering Assessment (Section 4 of NEI 00-04)
3. Component Safety Significance Assessment (Section 5 of NEI 00-04)

4. Defense-In-Depth Assessment (Section 6 of NEI 00-04)
5. Preliminary Engineering Categorization of Functions (Section 7 of NEI 00-04)
6. Risk Sensitivity Study (Section 8 of NEI 00-04)
7. IDP Review and Approval (Section 9 of NEI 00-04)
8. SSC Categorization (Section 10 of NEI 00-04)

The licensee stated in Section 3.1.1 of the LAR that it will implement the risk-informed categorization process in accordance with NEI 00-04, Revision 0, as endorsed in RG 1.201, Revision 1.

The licensee provided further discussion of specific elements within the 10 CFR 50.69 categorization process that are delineated in the NEI 00-04, Revision 0, as endorsed by RG 1.201, Revision 0. Elements of the licensee's categorization process are provided below. A more detailed review of those specific elements in the categorization process is discussed in this SE.

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

- Qualitative Characterization. System functions are categorized qualitatively as HSS or LSS based on the seven questions in Section 9.2 of NEI 00-04, Revision 0.

- Cumulative Risk Sensitivity Study. For PRA-modeled components, an overall risk sensitivity study is used to confirm that the population of LSS components results in acceptably small increases to CDF and LERF and meets the risk acceptance guidelines of RG 1.174, Revision 3.

- Review by the IDP. The categorization results are presented to the IDP for review and approval. The IDP reviews the categorization results and makes the final determination on the safety-significance of system functions and components.

Further, 10 CFR 50.59(c)(1)(v) requires that the SSC categorization be performed for entire systems and structures, not for selected components within a system or structure. Section 2 of the NEI 00-4 provides guidance of meeting this requirement.

The regulatory requirements in 10 CFR 50.69 and 10 CFR Part 50, Appendix B, and the performance monitoring outlined in NEI 00-04, Revision 0, as endorsed by RG 1.201, Revision 1, ensure that the SSC categorization process determines the safety significance of SSCs and categorizes them into one of four RISC categories. The licensee's SSC categorization program includes the appropriate steps/elements prescribed in NEI 00-04, Revision 0 to ensure that SSCs are appropriately categorized consistent with 10 CFR 50.69. Based on the above and as discussed throughout this SE, the NRC staff concludes that the proposed change meets the first key principle for risk-informed decision-making prescribed in RG 1.174, Revision 3.

3.2.2 Key Principle 2: LB Change is Consistent with the Defense-In-Depth Philosophy

Defense-in-depth (DID) is an approach to designing and operating nuclear facilities that prevents and mitigates accidents that release radiation or hazardous materials. The key is creating multiple independent and redundant layers of defense to compensate for potential human and mechanical failures so that no single layer, no matter how robust, is exclusively relied upon. DID includes the use of access controls, physical barriers, redundant and diverse key safety functions, and emergency response measures.

As discussed in RG 1.174, consistency with the defense-in-depth philosophy is maintained by the following considerations:

- Preserve a reasonable balance among the layers of defense.
- Preserve adequate capability of design features without an overreliance on programmatic activities as compensatory measures.
- Preserve system redundancy, independence, and diversity commensurate with the expected frequency and consequences of challenges to the system, including consideration of uncertainty.
- Preserve adequate defense against potential common-cause failures.
- Maintain multiple fission product barriers.
- Preserve sufficient defense against human errors.
- Continue to meet the intent of the plant's design criteria.

This principle exists in the 10 CFR 50.69 context, through subsection 50.69(c)(1)(iii) which requires the SSC categorization process to "[m]aintain [DID]."

Section 6 of NEI 00-04, Revision 0, provides guidance on DID assessment in the 10 CFR 50.69 context.

Figure 6-1 in NEI 00-04, Revision 0, provides guidance to assess design basis DID based on the frequency of the design-basis internal initiating event and the number of redundant and diverse trains nominally available to mitigate the initiating event. For each initiating event frequency, components are assigned as HSS if fewer than the indicated number of mitigating trains are nominally available. Section 6 of NEI 00-04, Revision 0, 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.

RG 1.201, Revision 1, endorses the guidance in Section 6 of NEI 00-04 but notes that the containment isolation criteria in this section of the guidance, 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. The 10 CFR 50.69(b)(1)(x) criteria are not used to determine the proper RISC category for containment isolation valves or penetrations.

In Section 3.1.1 of the LAR, the licensee clarifies that it will require an SSC to be categorized as HSS based on the DID assessment performed in accordance with NEI 00-04, Revision 0. Therefore, the NRC staff finds that the licensee's process is consistent with the NRC endorsed guidance in NEI 00-04, and therefore, fulfills the 10 CFR 50.69 (c)(1)(iii) criterion that requires

DID to be maintained. It also satisfies the second key principle for risk-informed decisionmaking in RG 1.174, Revision 3.

3.2.3 Key Principle 3: LB Change Maintains Sufficient Safety Margins

In risk-informed decisionmaking, the guidance in RG 1.174, Revision 3, describes the expectation that licensing basis changes "maintain[] sufficient safety margins." In the 10 CFR 50.69 context, this is ensured through the regulation in 10 CFR 50.69(c)(1)(iv) which requires, in part, reasonable confidence that sufficient safety margins are maintained for SSCs categorized as RISC-3.

The engineering evaluation that will be conducted by the licensee under 10 CFR 50.69 for SSC categorization will assess the design function(s) and risk significance of the SSC to assure that sufficient safety margins are maintained. The guidelines used for making that assessment will include ensuring that the categorization of the SSC does not adversely affect any assumptions or inputs to the safety analysis; or, if such inputs are affected, justification is provided to ensure sufficient safety margin will continue to exist.

The SSCs' design basis function as described in the plants' licensing bases, including the Updated Final Safety Analysis Report and TS Bases do not change and should continue to be met. Similarly, there is no impact to safety analysis acceptance criteria as described in the plant licensing basis.

Sufficient safety margin will be maintained because the RISC-3 SSCs will remain capable of performing their safety-related functions as required by 10 CFR 50.69(d)(2), and because any potential increases in CDF and LERF that might stem from changes in RISC-3 SSC reliability due to reduced treatment permitted by 10 CFR 50.69 will be small, as required by 10 CFR 50.69(c)(1)(iv). Therefore, the NRC staff finds that the program implemented by the licensee, consistent with the endorsed guidance in NEI 00-04, fulfills the 10 CFR 50.69(c)(1)(iv) criteria that sufficient safety margins are maintained. It also thus satisfies the third key principle for risk-informed decisionmaking prescribed in RG 1.174, Revision 3.

The system engineering assessment discussed in NEI 00-04, Section 4 and described below is an integral part of the evaluation that will be conducted by the licensee under 10 CFR 50.69 for SSC categorization to assess the design function(s) of the SSCs in order to assure that sufficient safety margins are maintained.

Paragraph 50.69(c)(1)(ii) of 10 CFR requires licensees to determine SSC functional importance using an integrated, systematic process for addressing initiating events (internal and external), SSCs, and plant operating modes, including those not modeled in the plant-specific PRA. The functions to be identified and considered include design basis functions and functions credited for mitigation and prevention of severe accidents. Section 4 of Revision 0 of NEI 00-04 provides guidance for developing a systematic engineering assessment involving the identification and development of base information necessary to perform the risk-informed categorization.

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

Section 4 of NEI 00-04 states that system selection and boundary definition include defining system boundaries where the system interfaces with other systems.

The guidance in NEI 00-04 also includes consideration of interfacing functions. Section 4 of NEI 00-04 provides guidance for circumstances when "the categorization of a candidate [LSS] SSC within the scope of the system being considered cannot be completed because it also supports an interfacing system." The guidance states in part:

> [i]n this case, the SSC will remain uncategorized until the interfacing system is considered […]. Therefore, the SSC will remain uncategorized and continue to receive its current level of treatment requirements.

Section 7.1 of NEI 00-04 states in part:

> Due to the overlap of functions and components, a significant number of components support multiple functions. In this case, the SSC or part thereof should be assigned the highest risk significance for any function that the SSC or part thereof supports.

In Section 2.2 of the LAR, the licensee stated:

> The safety functions 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. In Section 3.1.1 of the LAR, the licensee confirmed 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). Therefore, the NRC staff finds that the process described in the LAR is consistent with NEI 00-04, Revision 0, as endorsed by the NRC in RG 1.201, Revision 1, and meets the requirements set forth in paragraph 50.69(c)(1)(ii).

3.2.4 Key Principle 4: Change in Risk is Consistent with the Safety Goals

Key Principle 4 is centered on risk considerations, namely that any increase in risk resulting from the proposed licensing basis change is small and consistent with the intent of the Commission's policy statement on safety goals for the operations of nuclear power plants.

The risk-informed considerations prescribed in NEI 00-04, Revision 0, endorsed by RG 1.201, Revision 1, address the fourth and fifth key principles of the staff's standards for risk-informed decisionmaking, pertaining to the assessment for change in risk and monitoring the impact of the LB change.

The licensee addressed the above provisions provided in the applicable guidance. A summary of how the licensee's SSC categorization process is consistent with the guidance and methodology prescribed in NEI 00-04, Revision 0, and RG 1.201, Revision 1, is provided in the sections below.

3.2.4.1 Assembly of Plant-Specific Inputs

Paragraph 50.69(c)(1)(ii) of 10 CFR requires licensees to determine SSC functional importance using an integrated, systematic process for addressing initiating events (i.e., internal and external), SSCs, and plant operating modes, including those not modeled in the plant-specific

PRA. The functions to be identified and considered include design-basis functions and functions credited for mitigation and prevention of severe accidents.

Section 3 of NEI 00-04, Revision 0, 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. This step also includes the critical evaluation of plant-specific risk information to ensure that the PRA is modeled adequately to support the risk characterization of the SSCs for this application.

The NRC staff acknowledges that elements of the categorization process are not always performed in chronological order and may be performed in parallel, such that, the systematic process for evaluating the plant-specific PRA may include other aspects of the categorization process (e.g., system selection, system boundary definition, identification of system functions, and mapping of components to functions). The NRC staff's review of the PRAs and non-PRA methods is provided in Sections 3.2.4.3 and 3.2.4.4 of this SE.

In Section 3.1.1, "Overall Categorization Process," of the LAR, the licensee stated, in part, that:

> SNC will implement the risk categorization process in accordance with NEI 00-04, Revision 0, …as endorsed by Regulatory Guide (RG) 1.201 [Revision 1].

The NRC staff finds that the process described in the LAR and the guidance and summarized above is consistent with, and capable of, collecting and organizing information at the system level by defining boundaries, functions, and components and is, therefore, acceptable.

3.2.4.2 Component Safety Significance Assessment

Section 5 of NEI 00-04, Revision 0 provides guidance on the component safety significance assessment. This assessment assesses the safety-significance of components using quantitative or qualitative risk information from a modeled PRA hazard or other risk assessment method. In the NEI 00-04 guidance, component risk significance is assessed separately for the following hazard groups:

- Internal Events and Internal Floods
- Fire Events
- Seismic Events
- Other External Hazards (e.g., high winds, external floods)
- Shutdown Events

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. This paragraph of the rule further specifies that the PRA used in the categorization process must be of sufficient quality and level of detail and subject to an acceptable peer review process.

For hazards other than internal events, 10 CFR 50.69(b)(2) allows, and the guidance in NEI 00-04, Revision 0, summarizes, the use of PRA, if such PRA models exist, or, in the absence of quantifiable PRA, the use of other methods (e.g. FIVE, SMA, and shutdown safety management plan).

In Sections 3.1.1 and 3.2.1 through 3.2.3 of the LAR, the licensee described that the Hatch categorization process uses PRA modeled hazards to assess risks from internal events, internal flooding, internal fire, and seismic events. For the other risk contributors, the licensee's process uses the following non-PRA methods to characterize the risk:

- Non-seismic External Hazards and Other Hazards: Screening analysis performed using Part 6 of ASME/ANS RA-Sa-2009, as endorsed by the NRC.

- Shutdown Events: Safe Shutdown Risk Management program consistent with NUMARC 91-06.

- Passive Components: Arkansas Nuclear One, Unit 2 (ANO-2), passive categorization methodology (Reference 16).

The approaches and methods proposed by the licensee to address internal events, fire, seismic, other external hazards, and shutdown events are consistent with the approaches and methods included in the guidance in NEI 00-04, Revision 0, as endorsed in RG 1.201, Revision 1. The method for the categorization of passive components is an alternative method not specified in the NEI 00-04 guidance but is consistent with the plant-specific approval of ANO-2 for passive components for risk-informed safety classification and treatment for repair/replacement activities in Class 2 and Class 3 moderate and high energy systems. The NEI 00-04 guidance, as endorsed by RG 1.201, Revision 1, considers the results and insights from the plant-specific PRA peer reviews as required by 10 CFR 50.69(c)(1)(i), and non-PRA risk characterization as required by 10 CFR 50.69(c)(1)(ii).

The NRC staff's review of the information provided by the licensee for the modeled PRA hazards and non-PRA methods for acceptability is provided in the following subsections, 3.2.4.3 and 3.2.4.4 of this SE.

3.2.4.3 Evaluation of PRA Acceptability to Support the SSC Categorization Process

As noted in the previous section, consistent with Section 5 of NEI 00-04, Revision 0, the component safety significance assessment must include evaluations for each of the hazards' contribution to plant risk: (1) internal events hazard, (2) internal fire hazard, (3) seismic hazard, (4) other hazards (e.g., high wind, external floods, etc.), and (5) shutdown events.

Paragraph 50.69(c)(1)(i) of 10 CFR requires, in part, that a licensee's PRA must be of sufficient quality and level of detail to support the categorization process and must be subject to a peer review process assessed against a standard or set of acceptance criteria that is endorsed by the NRC. Section 50.69(b)(2)(iii) of 10 CFR requires the results of the peer review process conducted to meet 10 CFR 50.69(c)(1)(i) criteria be submitted as part of the application.

3.2.4.3.1 Scope of the PRA

The Hatch PRA is comprised of a full power, Level 1, internal events PRA (IEPRA), internal flooding PRA (IFPRA), internal fire PRA (FPRA), and seismic PRA (SPRA), all of which evaluate the CDF and LERF risk metrics.

For this LAR, SNC stated that the same PRA models were used for the separate LAR submitted in letter on April 4, 2018, for transition to the National Fire Protection Association Standard 805 (NFPA-805) (Reference 17) and requested the NRC staff coordinate its review of the PRA acceptability for the IEPRA and FPRA for both applications. Furthermore, in Section 3.3 of the LAR, the licensee confirmed that the IEPRA, IFPRA, FPRA, and SPRA have all been assessed against RG 1.200, Revision 2 consistent with NRC Regulatory Issue Summary (RIS) 2007-06 (Reference 18) . In evaluating the IEPRA, IFPRA, FPRA, and SPRA, the NRC staff considered: (1) peer review history, (2) the Appendix X, Independent Assessment process, (3) credit for FLEX in the PRA(s), and (4) assessment of assumptions and approximations. The staff reviews of these aspects are provided in subsections 3.2.4.3.2 through 3.2.4.3.7 of this SE.

3.2.4.3.2 Internal Events PRA and Internal Floods Peer Review History

In LAR Section 3.3, as supplemented in response to RAI 01.a, the licensee stated that a full scope peer review of the IEPRA and IFPRA was performed in November 2009. As stated in the LAR for transition to the NFPA Standard 805 this peer review was performed against RG 1.200, Revision 2, and used the process documented in NEI 05-04 (Reference 19). Furthermore, a facts and observations (F&O) closure was performed in April 2017 (Reference 20) using the process documented in Appendix X to NEI 05-04 (Reference 19), NEI 07-12 (Reference 21)and NEI 12-13 (Reference 22), “Close-Out of Facts and Observations” as accepted by the NRC in the letter dated May 3, 2017 (Reference 23). Review of the licensee’s Appendix X F&O Closure process is provided in Section 3.2.4.3.5 of this SE.

In LAR Attachment 3, the licensee provided dispositions to four finding-level F&Os that remained open after the F&O closure review process. For each F&O, the licensee provided a basis for why the uncompleted resolution of the findings do not impact the application. The NRC staff evaluated each disposition presented in Attachment 3 for significant impacts to the application. For three of the F&Os (i.e., F&Os 1-9, 1-15, and 6-8), the licensee provided sufficient information for NRC staff to find that resolution of these open F&Os would not impact the 10 CFR 50.69 application.

Internal flooding F&O 4-5 related to supporting requirement IFSN-A10 and IFQU-A5 identified that no credit is taken for the manual isolation of floods. The independent assessment team determined that this F&O identifies a major modeling issue with internal flooding, and therefore, constitutes significant changes from the previously peer reviewed model. In PRA RAI 02 (Reference 6) the NRC staff requested that the licensee confirm whether the IFPRA credits manual isolation of internal flooding and whether this modeling change had been peer reviewed. In response to PRA RAI 02 (Reference 2) the licensee stated that manual isolation of internal flooding was credited in the IFPRA and that the modeling, which resolved F&O 4-5, was determined to be a PRA upgrade in accordance with the ASME/ANS PRA standard. In RAI response letter dated December 18, 2019 (Reference 24), the license confirmed that a focused-scope peer review was performed in July 2019 to address F&O 4-5 and F&O 1-15 pertaining to documentation of flooding evaluations into the final notebook. The licensee stated that these two F&Os were resolved and no new finding-level F&Os were issued. The licensee stated that in 2018 the IFPRA model was separated from the IEPRA model and the IFPRA was updated to update the HRA analysis of the flooding isolation actions. The licensee also confirmed that the 10 CFR 50.69 categorization process will use the IFPRA the model of record. Separation of the internal floods model from the internal events model does not impact the NRC staff’s review of the IEPRA and IFPRA for acceptability.

The NRC staff concludes that the licensee has followed the guidance in RG 1.200, submitted the results of the peer reviews, and appropriately dispositioned the F&Os to assess the impact on the risk-informed application. Therefore, the NRC staff finds that the Hatch IEPRA and IFPRA were appropriately peer reviewed consistent with RG 1.200, Revision 2, and therefore, meet the requirement set forth in in 10 CFR 50.69(b)(2)(iii). Further, the NRC staff finds that the quality and level of detail of the IEPRA and IFPRA is sufficient to support the categorization of SSCs using the process endorsed by the NRC staff in RG 1.201. and therefore, meets the requirement in 10 CFR 50.69(c)(1)(i).

3.2.4.3.3 Fire PRA Peer Review History

The last full-scope peer review of the FPRA was performed in May 2016. As stated in the LAR to allow Hatch to transition to the NFPA Standard 805 this peer review was performed against the ASME/ANS 2009 Standard, as endorsed by RG 1.200, Revision 2. Furthermore, the licensee discussed that an Appendix X F&O closure was performed in October 2017 using the process documented in Appendix X to NEI 05-04 (Reference 19), NEI 07-12 (Reference 21) and NEI 12-13 (Reference 22), "Close-Out of Facts and Observations" as accepted by the NRC in a letter dated May 3, 2017 (Reference 23). A detailed staff review of the Appendix X process in general is discussed in Section 3.2.4.3.5 of this SE.

In response to RAI 01.a, the licensee stated that resolution of one Finding-level F&O (i.e., F&O 20-18) was determined by the F&O closure review team to constitute a PRA upgrade. The licensee stated that a focused-scope peer review was performed concurrent with the F&O Closure to review the licensee's method of calculating time-to-cable-damage due to exposure to fire and that the IA team determined the method was technically sound and provided a reasonable and realistic estimate. The licensee also stated that no additional F&Os were issued as a result of the focused-scope peer review.

The staff review of the FPRA for 50.69 application is based on the review performed for the NFPA-805 LAR, as supplemented, where the NRC staff conducted a detailed technical review of the FPRA. In that SE (Reference 25), the NRC staff concluded that (1) the FPRA model adequately represents the current, as-built, as-operated, and as-maintained plant, as it will be configured after full implementation of NFPA-805; (2) the FPRA model conforms sufficiently to the applicable industry PRA standards at an appropriate capability category; and (3) the fire modeling used to support the development of the FPRA is appropriate and acceptable.

Attachment 1 of the LAR listed proposed NFPA-805 plant modifications that are required to be completed in order to meet the acceptance criteria of RG 1.174. Furthermore, the FPRA risk estimates presented in LAR Attachment 2 reflect the plant after the implementation of the modifications in Attachment 1. Since the rule in 10 CFR 50.69(c)(1)(ii) requires that the PRA reflect the current plant configuration and operating practices and applicable plant and industry operational experience, the NRC staff requested in RAI 03 (Reference 6) that the licensee confirm or provide a mechanism to ensure that the FPRA model used for SSC categorization reflects the as-built, as-operated, and as-maintained plant; meets RG 1.174 acceptance requirements; and incorporates any FPRA updates needed to resolve RAIs from the NFPA-805 review for the FPRA acceptability. In response to RAI 03 and RAI 12, the license stated that the FPRA model changes that were necessary to resolve questions associated with the staff review for the NFPA-805 LAR have been incorporated. The licensee proposed to add text to the license condition, as further discussed in Section 4.0 of this SE. The license condition ensures that: (1) the FPRA that will be used in 10 CFR 50.69 categorization will reflect the as-built and as-operated, plant; and (2) the licensee will confirm that the RG 1.174, Revision 3 risk

acceptance guidelines are met prior to implementing the 10 CFR 50.69 categorization process. The staff notes that while the proposed NFPA-805 plant modifications are implemented, meeting the RG 1.174, Revision 3 risk acceptance guidelines may require future changes to the PRA. Such changes are an expected part of the normal PRA periodic review and configuration control and therefore captured in paragraphs (e) and (f) of 10 CFR 50.69 to ensure the PRA continually reflects the as-built, as-operated, and as-maintained plant. A more detailed staff review of key principle 5, which governs monitoring the impact of the proposed change is reviewed in Section 3.2.5 of this SE.

The NRC staff finds that the quality and level of detail of the FPRA is sufficient to support the categorization of SSCs as required by 10 CFR 50.69(b)(2)(ii) and uses the process endorsed by the NRC staff in RG 1.201. The NRC staff's conclusion is based on the following: (1) the staff review of the FPRA acceptability for NFPA-805 is applicable to the 10 CFR 50.69 risk-informed application; (2) licensee's confirmation that all FPRA model changes necessary to resolve the staff's RAIs associated with the NFPA-805 LAR review have been incorporated into the FPRA model; (3) the licensee's proposed license condition requiring confirmation that the RG 1.174, Revision 3, risk acceptance guidelines are met prior to implementing 10 CFR 50.69 categorization process, and (4) the licensee established a periodic review and programmatic configuration control process, as reviewed in Section 3.2.5 of this SE, to incorporate plant and PRA changes into the categorization processes and to ensure that current SSC categorization(s) remains valid. The NRC staff concludes that the quality of the FPRA, with the proposed license condition, meets the requirement in 10 CFR 50.69(c)(1)(i).

3.2.4.3.4 Seismic PRA

The NRC staff's review of the SPRA was based on the results of the peer review of the SPRA and a subsequent F&O closure review as described in LAR Sections 3.2.3 and 3.3 and supplemental information. The last full-scope peer review of SPRA was performed in October 2016 against the ASME/ANS RA-Sb-2013 PRA standard (Reference 26). The subsequent F&O closure review was performed in June 2017 and, as a result, all Finding-level F&Os were closed.

In its letter dated March 7, 2018 (Reference 27), the NRC staff accepted, with clarifications and qualifications, NEI 12-13, which provides industry peer review guidance for external hazard PRAs, including SPRAs. In RAI 09 (Reference 6) the NRC staff requested discussion of how the SPRA peer reviews considered the NRC staff comments in its letter dated March 7, 2018. In the response to RAI 09 in its supplement dated July 16, 2019, the licensee provided dispositions to the 32 comments from the NRC staff on NEI 12-13 in the March 7, 2018, letter. NRC staff reviewed the licensee's dispositions and finds that the licensee's SPRA peer review was consistent with the NRC staff's acceptance of NEI 12-13. The NRC staff's comments were addressed appropriately as part of the peer review.

In RAI 08, NRC staff noted that the SPRA was peer reviewed using the requirements in Addendum B of the ASME/ANS RA-Sb-2013 PRA standard which is not endorsed by the NRC, as noted in the July 6, 2011, letter (Reference 28). The NRC staff requested a plant specific evaluation to the SRs in Addendum A for Hatch. In response to RAI 08, the licensee stated that the comparison between the surveillance requirements (SRs) for SPRA (i.e., Part 5 of the PRA standard) in Addenda A and B included in "Response to Supplemental Information Needed for Acceptance of Systematic Risk-Informed Assessment of Debris Technical Report" (Reference 29), was applicable to the Hatch SPRA. The licensee also provided a table dispositioning Addendum B SRs, SHA-B3, SHA-C3, SFR-C3 SFR-C6, SFR-G3, and SPR-B1, that did not

equate to the Addendum A SRs or envelope them. For these SRs, the licensee provided justification that the Hatch SPRA is in conformance with the corresponding Addendum A SRs with one exception. For SR SFR-C6, the licensee stated that the Hatch SPRA conforms to current state-of-practice and uncertainties in the soil-structure interaction analysis for the Hatch SPRA and were accounted for by using soil properties derived from probabilistic evaluation via the local site response analysis which includes epistemic and aleatory uncertainties. Based on its review of (1) the comparison of SRs in Part 5 of Addendum B of the ASME/ANS 2009 Standard to those in Addendum A, incorporated by Reference by the licensee, and (2) the supplemental information which provided details of the licensee's approach for specific SRs including SFR-C6, the NRC staff finds that the licensee's use of Addendum B adequately addresses the technical elements for the development of an SPRA. Therefore, the NRC staff concludes that the analysis as provided in response to the RAIs supporting use of Addendum B is an acceptable plant-specific alternative to the NRC-endorsed approach. This SE does not provide generic approval Addendum B beyond the Hatch plant-specific SPRA used to support this application.

Based on the above, 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). Further, the NRC staff finds that the quality and level of detail of the SPRA is sufficient to support the categorization of SSCs using the process endorsed by the NRC staff in RG 1.201 and, therefore, meets the requirement in 10 CFR 50.69(c)(1)(i).

3.2.4.3.5 Appendix X, Independent Assessment Process for F&O Closure

Section X.1.3 of Appendix X to NEI 05-04, NEI 07-12, and NEI 12-13, as accepted by the NRC staff in memorandum dated May 3, 2017, provides guidance on how to perform an independent assessment for the closure of F&Os resulting from a peer review. The independent assessment process, described in Appendix X, includes criteria for: (1) the qualifications of independent assessment team members, (2) pre-review activities, (3) on-site review activities, and (4) post-review activities. The Appendix X process assures that closure of the F&Os are met at Capability Category (CC) II for the applicable supporting requirements (SRs) in the ASME/ANS RA-Sa-2009 PRA Standard as endorsed by RG 1.200 Revision 2.

In LAR Section 3.3, the license stated that resolutions to IEPRA, FPRA, and SPRA peer review finding-level F&Os were reviewed and closed using the process documented in Appendix X. Section X.2 of the Appendix X guidance states, in part, "[o]nce an F&O is closed out, the utility is not required to present and explain them in peer reviews, NRC submittals or other requests excluding NRC audits."

In RAI 01, NRC staff requested information about how the identification of PRA upgrades and the potential need for a focused-scope peer review was determined during the F&O closure process for the IEPRA, FPRA, and SPRA to confirm those reviews were performed in accordance with guidance in Appendix X and the conditions stipulated in the NRC acceptance letter dated May 3, 2017. In response to RAI 01.a, the licensee described that for each F&O process (conducted for the IEPRA, FPRA, and SPRA), the Independent Assessment (IA) team was provided with a characterization, for each finding-level F&O resolution, as a PRA upgrade or maintenance update, using guidance consistent with the ASME/ANS RA-Sa-2009 PRA standard. The licensee's use of the definition of a PRA upgrade from the ASME/ANS PRA standard is consistent with the guidance provided in Appendix X as accepted by the NRC in its letter dated May 3, 2017. Therefore, the NRC staff finds that the IA team appropriately

assessed each F&O to determine if it constituted a PRA upgrade or update consistent with the Appendix X process, as accepted by the NRC.

In response to RAI 01.a and 01.c (Reference 2), the licensee stated the IA team determined that resolution of one finding-level FPRA F&O (i.e., F&O 20-18) constituted a PRA upgrade. As a result, a concurrent focused-scope peer review was performed to review the PRA change that calculated the time-to-cable-damage due to exposure to a fire. The licensee stated that the IA team determined that the method was technically acceptable because it provided a reasonable and realistic method for estimating time to cable damage due to exposure in a fire environment. The licensee confirmed that no additional F&Os were generated as a result of the focused-scope peer review.

In further response to RAI 01.c, for the SPRA, the licensee stated that the IA team determined that the resolution of two SPRA Finding-level F&Os (6-2 and 6-10) constituted a PRA upgrade. As a result, a concurrent focused-scope peer review was performed, and no additional F&Os were issued as a result of this focused-scope peer review.

Appendix X guidance states in part, "[t]he relevant PRA documentation should be complete and have been incorporated into the PRA model and supporting documentation prior to closing the finding." For closure of F&O(s) after the on-site review, Appendix X guidance explicitly states, "[t]he host utility may, in the time between the on-site review and the finalization of the independent assessment team report, demonstrate that the issue has been addressed, that a closed finding has been achieved, and that the documentation has been formally incorporated in the PRA Model of Record."

As part of RAI 01, the staff requested the licensee confirm that all model changes associated with the closure of all F&Os reviewed during the IA performed in May 2017 were incorporated into the PRA and/or the supporting documentation at the time of the finalization of the IA team report. In response to RAI 01, the licensee confirmed that the SPRA model of record includes all changes associated with the closure of all finding- level F&Os reviewed during the IA performed in May 2017. As discussed above, RG 1.200, Revision 2, does not provide specific guidance on the PRA configuration control for a specific risk-informed application. However, it does acknowledge that application-specific PRA models exists. Section 3.2 of the NEI 00-04 guidance describes, in part, "[a]n essential element of the SSC categorization process is a plant-specific full power internal events PRA, which should ... reflect the as-built and as-operated plant..... " The living PRA model includes the maintenance updates incorporated into the plant to represent the as-built, as-operated, and as-maintained plant. Further, as described in Section 4.0 of this SE, a part of the license condition for implementation of the 10 CFR 50.69 SSC categorization process provided in Enclosure 2 of the letter dated December 18, 2019, SNC committed to update the PRA models (i.e., IEPRA, IFPRA, FPRA, and SPRA) to reflect the as-built, as-operated plant. Furthermore, paragraph 50.69(f) for programmatic change control requires the licensee to document the basis for its categorization of any SSC under paragraph 50.69(c) of 10 CFR before removing any requirements under 10 CFR 50.69(b)(1) for those SSCs. Paragraph 50.69(c)(i) of 10 CFR requires in part, "[t]he 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 endorsed by the NRC." A more detailed staff review of the IEPRA, IFPRA, FPRA, and SPRA peer review history is provided in Sections 3.2.4.3.2 through 3.2.4.3.4 of this SE.

Based on the NRC staff’s review of the licensee's responses to RAI 01 the NRC staff concludes that the F&Os for the IEPRA, IFPRA, FPRA, and SPRA have been appropriately assessed by the IA team and therefore have been addressed adequately for the SSC categorization process.

3.2.4.3.6 Credit for FLEX Equipment

In RAI 06, the NRC staff requested that the licensee confirm whether FLEX equipment and actions were credited in the IEPRA, FPRA and SPRA, and, if credited, provide additional information on how the FLEX equipment and actions are modeled, and justification that crediting FLEX in the PRA did not represent a PRA upgrade. In response to RAI 06, the licensee confirmed that certain FLEX equipment installed as permanent plant equipment is credited in the IEPRA, FPRA and SPRA models. The licensee stated that plant procedures have been revised to include operation of this permanently installed equipment. The licensee explained that the permanent FLEX equipment being credited consists of new panels providing power to critical instrumentation cabinets, manual transfer switches, and modifications to add a backup air supply to the hardened containment vent system. The licensee explained that this equipment (i.e., instrument panels, inverters, air accumulators, manual valves, check valves) is very similar to other plant equipment modeled in the PRAs and that there is sufficient plant-specific and industry generic data to support the modeling of this equipment. The licensee stated that no FLEX actions outside of the main reactor buildings or control building are credited in the PRA models and that the credited operator actions have been peer reviewed and are similar to other operator actions evaluated using approaches consistent with the ASME/ANS PRA Standard as endorsed by RG 1.200 Revision 2. The licensee further stated that the modeling of the credited FLEX equipment and actions (1) does not represent any new methods, (2) does not change the scope of the model(s) given the equipment, dependencies, and type of accident sequences remain the same or (3) does not represent a change in capability of the PRA model given the original and updated models can both evaluate the risk associated with loss-of-offsite power and station blackout. Thus, the licensee confirmed that the changes implemented for the incorporation of the FLEX modifications were within the framework of the existing peer reviewed PRA model structure. The NRC concludes that the licensee assessed the plant-specific incorporation of the FLEX equipment and actions consistent with the definition of a PRA upgrade as described in the ASME/ANS PRA Standard RA-Sa-2009, therefore, the NRC staff finds that the FLEX modeling does not represent a PRA upgrade requiring a focused-scope peer review.

The NRC staff finds that the licensee’s modeling of FLEX is acceptable because the licensee only credits its permanently installed FLEX equipment that is similar to other plant equipment already modeled in the PRAs (i.e., sufficient plant-specific and industry generic data exist for modeled equipment). Moreover, no FLEX actions outside of the main reactor buildings or control building were credited in the PRA models.

3.2.4.3.7 Assessment of PRA Key Assumptions and Sources of Uncertainty

Paragraphs 50.69(c)(1)(i) and (ii) of 10 CFR require that a licensee’s PRA be of sufficient quality and level of detail to support the 10 CFR 50.69 categorization process and 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 operational experience.

Section 3.3.2 of RG 1.200, Revision 2, provides guidance that states, in part, "[f]or each application that calls upon this regulatory guide, the applicant identifies the key assumptions and approximations relevant to that application. This will be used to identify sensitivity studies as input to the decision-making associated with the application."

Section 4.2, of RG 1.200, Revision 2 provides guidance that identifies specific information to be included in the licensee submittal to demonstrate the technical adequacy of the PRA used in an application is of sufficient quality. The identified information includes the identification of the key assumptions and approximations relevant to the results used in the decision-making process.

Table A-1 of RG 1.200, Revision 2, entitled, "Staff Position on ASME/ANS RA-Sa-2009 Part 1, General Requirements for an At-Power Level 1 and LERF PRA", includes a staff clarification for Section 1-6.1 of ASME/ANS RA-Sa-2009. The resolution for this clarification states, in part, "the peer review shall also assess the appropriateness of the assumptions."

Regulatory Guide 1.174, Revision 3 (Reference 11), cites NUREG-1855, Volume 1 (Reference 30), as related guidance that includes changes associated with expanding the discussion of sources of uncertainties. NUREG-1855, Revision 1 explicitly states, in part, "[RG] 1.200 . . . and the PRA consensus standard published by ASME and [ANS] . . . each recognize the importance of identifying and understanding uncertainties as part of the process of achieving acceptability in a PRA, and these References provide guidance on this subject." Revision 1 of NUREG-1855 identifies EPRI TR-1026511, "Practical Guidance on the Use of Probabilistic Risk Assessment in Risk-Informed Applications with a Focus on the Treatment of Uncertainties" (Reference 31) and EPRI TR 1016737, "Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments," (Reference 32) as providing additional guidance for identifying and characterizing key sources of uncertainty.

Process for Identification of Key Assumptions and Sources of Uncertainty

Section 3.2.7 of the LAR stated that guidance in NUREG-1855, Volume 1, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making," (Reference 30) and Electric Power Research Institute (EPRI) TR-1016737, "Treatment of Parameter and Modeling Uncertainty for Probabilistic Risk Assessments" (Reference 32), was used to identify, characterize, and screen model uncertainties.

In RAI 04.a and 04.c (Reference 6), the NRC staff requested the licensee provide a description of the process and criteria used to identify the key assumptions and sources of uncertainty, consistent with the definitions provided in RG 1.200, Revision 2, and confirmation that the process is consistent with NUREG-1855, Revision 1 or other NRC-accepted guidance. In response to RAI 04.a (Reference 6), the licensee described the process and criteria used to identify key assumptions and sources of uncertainty for the IEPRA, IFPRA, FPRA, and SPRA models separately. The licensee also confirmed in response to RAI 04.c that the process it described in response to RAI 04.a is consistent with NUREG-1855, Volume 1 (Reference 30), NUREG-1855, Revision 0 (Reference 33), and NUREG/CR-6850 (Reference 34) and (Reference 35) for the different PRA models.

For the IEPRA and IFPRA, the licensee performed a review of assumptions made in the analysis for each technical element. If an assumption was judged to represent a potential source of model uncertainty for the technical element it was identified for additional consideration of its overall PRA impact. The licensee provided example criteria used to perform this identification that included: (1) the assumption introduced a conservative or

non-conservative bias in the results, (2) the assumption bases could not be directly identified as consistent with an industry consensus approach, and (3) the assumption is associated with phenomena, event, or failure mode being modeled was not completely understood. In addition, the licensee also considered the generic sources of uncertainty identified in EPRI TR-1016737, Appendix A, correlated with the SRs in the ASME/ANS PRA Standard. The licensee confirmed that, for the base IEPRA and IFPRA models, an updated review of modeling uncertainties was performed, using a process consistent with the guidance in NUREG-1855, Volume 1, including consideration of the implementation process in EPRI TR-1026511 and consideration of generic sources of uncertainty in EPRI TR-1016737.

For the FPRA, the licensee explained in response to RAI 04.a that it used a process to identify the FPRA model uncertainties that reflects the guidance in NUREG/CR-6850 (Reference 34), (Reference 35) (Reference 36), and EPRI-1016737, Appendix A (Reference 32). The licensee stated that it identified uncertainties associated with each of the FPRA tasks defined in NUREG/CR-6850. The process further evaluated assumptions made in each of the FPRA tasks to identify associated model uncertainties. The licensee stated that the identified key assumptions and sources of uncertainty are reasonably consistent with those presented EPRI TR-1026511, Appendix B, though the EPRI listing of uncertainties was not explicitly used.

For the SPRA, the licensee explained in response to RAI 04.a that it identified uncertainties associated with SPRA and provided sensitivity studies for those uncertainties. The licensee explained that, except for crediting portable FLEX equipment and the capacity-based screening, the sensitivity studies indicated that the SPRA was not overly sensitive to the other uncertainties and the seismic risk is a small percentage of the total risk. Therefore, the licensee stated that these uncertainties are unlikely to have a significant impact on this application and are not considered to be key sources of uncertainty for this application. The licensee also stated it performed an updated review of seismic modeling uncertainties using a process consistent with the guidance in NUREG-1855, Revision 1, and EPRI TR-1026511, Appendix C, though the listing of uncertainties provided in EPRI TR-1026511, Appendix C were not explicitly used. The licensee stated that important assumptions and sources of uncertainty from the internal events PRA, which is the foundation for the SPRA, were considered.

Treatment of the Key Assumptions and Sources of Uncertainty

Section 5 of the NEI 00-04, Revision 0, guidance as endorsed by the NRC states in part:

> An analysis of the impacts of parametric uncertainties on the importance measures used in this categorization process was performed and documented in EPRI TR-1008905, *Parametric Uncertainty Impacts on Option 2 Safety Significance Categorization* […]. The conclusion of this analysis was that the importance measures used in combination with identified set of minimum sensitivity studies adequately address parametric uncertainties.

Furthermore, the guidance in NEI 00-04, Revision 0 (Reference 8), specifies that sensitivity studies should be conducted for each PRA model to ensure that PRA assumptions and sources of uncertainty (e.g., human error, common cause failures (CCFs), and maintenance probabilities) do not mask the SSC's importance. Tables 5-2 and 5-3 of NEI 00-04 provides several recommended sensitivity studies for the IEPRA and FPRA models.
In the response to RAI 04, the licensee provided a list of updated key assumptions and sources of uncertainty along with the associated dispositions and impact on the SSC categorization process in revised Table 6-1 that supersede Table 6-1 in the LAR. For one key

assumption/uncertainty pertaining to impact of containment venting on the core cooling system net positive suction head (NPSH) for the disposition and impact to the risk-informed application, the licensee stated that the model is being revised to appropriately model the NPSH. The licensee discussed that upon the update to the model, NPSH will no longer be a modeling assumption or significant source of uncertainty and that the updated model will be used for SSC categorization. In review of the licensee’s response to RAI 18 for the transition to the NFPA 805, Performance-Based Standard (Reference 37), the licensee confirmed that the PRA model has been revised to remove the uncertainty associated with the assumed conditional probability for loss of NPSH for low pressure emergency core cooling system pumps following containment venting. The NRC staff concludes that upon review of the response provided in RAI 18 for staff review of NFPA-805, the licensee has updated the PRA model to remove the key assumption/source of uncertainty associated with the NPSH provided in revised Table 6-1, therefore it is no longer applicable as a key assumption/source of uncertainty for the SSC categorization process. The NRC staff reviewed the remaining key assumptions and sources of uncertainty for the IEPRA and IFPRA provided by the licensee in Table 6-1 (Update) and concluded that the licensee provided adequate dispositions for the impact on the risk-informed application.

In response to RAI 04.a, for the FPRA, the licensee provided four updated key assumptions and sources of uncertainty along with the associated dispositions for the application in an updated Table 6-3, which supersedes Table 6-3 from the LAR. In RAI 05.b, the NRC staff noted that the modeling conservatisms associated with untraced power cables in the FPRA can mask the importance measures of certain SSCs. The NRC staff then requested justification that this modeling uncertainty does not impact the application. In response to RAI 05.b, the licensee identified the systems with untraced cables, referred to in the calculations as cables with “unknown locations” (UNL). The licensee stated that these are primarily non-safety-related systems, mostly secondary-side systems, such as components associated with the main condenser, the condensate and feedwater systems, and the circulating water system. The licensee stated that the assumed unavailability of these systems (by not modelling them) results in overestimating the risk importance of other systems. Accordingly, if the UNL systems were credited, then the importance of other systems such as safety-related systems would decrease. Thus, the effect of not crediting UNL components is, in general, conservative because the safety significance of safety related components undergoing 10 CFR 50.69 categorization might be overestimated in some cases. To further justify this treatment, the licensee discussed that the dominant risk contributors in the FPRA are loss of offsite power caused by fire-induced loss of the startup transformers or the high voltage switchyard circuit breakers and fire-induced Loss-of-Coolant Accident caused by spurious opening of the Safety Relief Valves. For these scenarios, the licensee confirmed that the UNL components would not be available and therefore the treatment of the UNL items related to the importance of the modeled SSCs is bounded given the limited number of UNL component(s) and their plant function(s). The NRC staff finds that the licensee appropriately dispositioned this uncertainty item by establishing it has no impact on the application. The NRC staff reviewed the remaining updated key assumptions and sources of uncertainty provided in revised Table 6-3 for the FPRA and found the dispositions to be adequate for the application.

In response to RAI 04.a, the licensee provided an updated Table 6-2 that supersedes Table 6-2 from the LAR that lists two key assumptions and sources of uncertainty for the SPRA along with a disposition for each identified key assumption and source of uncertainty. One of these assumptions relates to the use of a capacity-based screening criterion to determine which SSCs are directly modeled in the SPRA. To address this key assumption, the licensee explained that its 10 CFR 50.69 categorization process would include a review of SSCs with fragilities greater

than the screening level to identify any seismic SSCs (or correlated group of SSCs) that could lead directly to core damage or large early release, and such SSCs would be addressed during the categorization process for the applicable system, consistent with NEI 00-04. The other key assumption concerns modeling of portable FLEX equipment. The licensee stated that portable FLEX equipment would not be credited in the SPRA model for the categorization of SSCs using the SPRA. The NRC staff's review finds that the licensee appropriately identified and dispositioned the key assumptions and sources of uncertainty in the SPRA for this application.

Based on the above, the NRC staff finds that the licensee appropriately searched for, identified, and evaluated key assumptions and sources of uncertainties for the IEPRA, IFPRA, FPRA and SPRA for this application consistent with the guidance provided in RG 1.200, NUREG-1855, Revision 1 and NEI 00-04, as endorsed by the NRC.

3.2.4.3.8 PRA Importance Measures and Integrated Importance Measures

Under 10 CFR 50.69(c)(1)(i) the SCC categorization process must, "[c]onsider results and insights from the plant-specific PRA." These requirements are met, in part, by using importance measures and sensitivity studies as described in the methodology in NEI 00-04, Section 5 and consistent with the ASME/ANS RA-Sa-2009 PRA standard, as endorsed in RG 1.200, Revision 2 (Reference 9).

RG 1.200, Revision 2, states, in part:

> Methods such as importance measure calculations (e.g., Fussell-Vesely Importance [(F-V)], risk achievement worth [(RAW)], risk reduction worth [(RRW)], and Birnbaum Importance) are used to identify the contributions of various events to the estimation of CDF for both individual sequences and the total CDF [i.e., both contributors to the total CDF, including the contribution from the different hazard groups and different operating modes (i.e., full- and low-power and shutdown) and contributors to each contributing sequence are identified].
>
> The results of the Level 2 PRA are examined to identify the contributors (e.g., containment failure mode, physical phenomena) to the model estimation of LERF or LRF [(Large Release Frequency)] for both individual sequences and the model as a whole […].

NEI 00-04, Revision 0, provides guidance where the F-V and RAW importance measures are obtained for each component, each PRA modeled hazard (i.e., separately for the IEPRA (including internal flood) and FPRA), and the values are then compared to specified criteria as follows:

> Components which have importance measure values that exceed the risk criteria (i.e., F-V greater than 0.005, RAW greater than 2, CCF RAW greater than 20) are assigned candidate safety-significant.

Section 5.1 of NEI 00-04, Revision 0, recommends that a truncation level of five orders of magnitude below the baseline CDF (or LERF) value should be used for calculating the F-V risk importance measures. The guidance also recommends that the truncation level used should be sufficient to identify all functions with a RAW value greater than 2.

Importance Measures

For the IEPRA, IFPRA, and FPRA the licensee can follow the guidance in NEI 00-04 described above regarding importance measures. In Section 3.1.1 of the LAR, the licensee stated:

> The IDP will be trained in the specific technical aspects and requirements related to the categorization process. Training will address ... the interpretation of risk importance measures ...

Regarding the SPRA, according to Section 5.3 of NEI 00-04, Revision 0, if a SPRA is being used to support the 10 CFR 50.69 categorization, then SSCs that have been screened out for seismic robustness should be retained to maintain validity of the final SSC importance determination. In RAI 07, NRC staff questioned whether the potential use of capacity-based screening used in the SPRA was consistent with the guidance in NEI 00-04. The NRC staff requested an explanation of whether the screening was used and, if so, said use maintains consistency with the importance measure criteria in NEI 00-04. In response to RAI 07, the licensee confirmed that it uses a screening level in the SPRA. The licensee also stated that when the fragility of an SSC was determined to be above the fragility screening value, no further refinement of the fragility was performed, but the SSC was included in the logic model when it was judged to be important to CDF and LERF. The licensee also stated that review of SSCs with capacities greater than the fragility screening level would be performed as part of the categorization to identify any SSC or correlated group of SSCs that could lead directly to core damage or large early release. The NRC staff's review finds that the licensee's screening of SSCs for seismic robustness is acceptable for this application because such components are included in the SPRA, and therefore, will be part of the categorization.

In RAI 11, the NRC staff requested an explanation of how importance measures were derived for the SPRA, given that the contribution of initiating events are discretized into bins. The NRC staff also requested justification of why the integrated derivation generated importance measures consistent with the NEI 00-04 guidance. In response to RAI 11, the licensee explained that the F-V and RAW measures for a component for each seismic acceleration interval were calculated using a weighted approach, and then the overall importance values (for F-V and RAW) for that component were determined by combining the importance values over all seismic acceleration intervals or "bins." The licensee further explained that the F-V for a component from the SPRA will be combined with the F-V of the random failures for that component from the SPRA to get a complete picture of the SPRA F-V importance measure for that component. In the case of RAW, the licensee stated that "the maximum of the RAW for seismically induced failure and RAWs of random failures for that component is used to get a complete picture of the SPRA RAW importance measure."

The NRC staff finds that the licensee's use and treatment of importance measures for the IEPRA, IFPRA, FPRA, and SPRA are consistent with the guidance in NEI 00-04, Revision 0, as endorsed in RG 1.201, Revision 1.

Integrated Importance Measures

Section 5.6 of NEI 00-04, Revision 0, titled, "Integral Assessment," discusses the need for an integrated computation using the available importance measures. The guidance further states, in part, that the, "integrated importance measure essentially weights the importance from each risk contributor (e.g., internal events, fire, and seismic PRAs) by the fraction of the total core

damage frequency [or large early release frequency] contributed by that contributor." The guidance also provides formulas to compute the integrated F-V and integrated RAW.

In LAR Section 3.1.1, the licensee stated that the process to categorize each system will be consistent with the guidance in NEI 00-04, as endorsed by RG 1.201. The NRC staff noted that the NEI guidance also provides formulas to compute the integrated F-V, and integrated RAW and the scope of modeled hazards for Hatch includes the IEPRA, IFPRA, FPRA, and SPRA.

The NRC staff finds that the licensee's approach to determine the integrated importance measures across each of these modeled PRA hazards is consistent with the guidance in NEI 00-04, Revision 0, as endorsed in RG 1.201, Revision 1. Therefore, the NRC staff finds the licensee's approach to determine the integrated importance measures across each hazard is acceptable.

3.2.4.3.9 PRA Acceptability Conclusions

Pursuant to 10 CFR 50.69(c)(1)(i), the categorization process must consider results and insights from a plant-specific PRA. The use of the IEPRA, FPRA and SPRA to support SSC categorization is endorsed by RG 1.201, Revision 1. Under 10 CFR 50.69(c)(1)(i), the PRAs "must be of sufficient quality and level of detail to support the categorization process and must be subjected to a peer review process assessed against a standard . . . that is endorsed by the NRC." Revision 2 of RG 1.200 provides guidance for determining the acceptability of the PRA by comparing the PRA against the relevant requirements of the ASME/ANS 2009 Standard using a peer review process. As discussed above, the licensee has subjected the IEPRA, IFPRA, FPRA, and SPRA to the peer-review processes and submitted the results of the peer review. The NRC staff reviewed the peer review history (which included the results and findings); the licensee's resolution of peer-review findings; the identification and disposition of key assumptions and sources of uncertainty; and the licensee's approach to importance measures. The NRC staff finds that (1) the licensee's IEPRA, FPRA, and SPRA are acceptable to support the categorization of SSCs using the process endorsed by the NRC staff in RG 1.201, Revision 1, and (2) the key assumptions and sources of uncertainty for the PRAs have been identified consistent with the guidance in RG 1.200, Revision 2 and NUREG-1855, as applicable, and addressed appropriately for this application.

Based on the above, the NRC staff finds the licensee provided the required information, and the IEPRA, IFPRA, FPRA and SPRA, meet the requirements set forth in 10 CFR 50.69(c)(1)(i) because the licensee's proposed process considers integrated importance measures, sensitivity studies, and uncertainty consistent with NEI 00-04, as endorsed by RG 1.201, Revision 1. The NRC staff similarly finds the licensee has complied with the 10 CFR 50.69(b)(2)(iii) requirement to submit the "[r]esults of the PRA process conducted to meet [10 CFR] 50.69(c)(1)(i)."

3.2.4.4 Evaluation of the Use of Non-PRA Methods in SSC Categorization

As required by 10 CFR 50.69(c)(1)(ii), the SCC categorization process must determine SSC functional importance using 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.

The licensee's categorization process uses the following non-PRA methods, respectively:

- Passive Components: ANO-2 passive categorization (Reference 16),
- Screening using criteria in Part 6 of the ASME/ANS RA-Sb-2013 PRA standard for non-seismic external hazards (e.g., high winds, external floods) and other hazards; (Reference 38),
- Safe Shutdown Risk Management program in accordance with NUMARC 91-06 (Reference 15).

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

3.2.4.4.1 Non-Seismic External Hazards and Other Hazards

In LAR Attachment 1, Section 3.2.4, the licensee indicated that it had reevaluated non-seismic external hazards and other hazards using the criteria in the ASME/ANS 2013 Standard. This hazard category includes all non-seismic external hazards such as high winds, external floods, transportation, and nearby facility accidents, as well as other external hazards. However, the NRC has not endorsed ASME/ANS RA-Sb-2013 (Reference 38). Therefore, in RAI 10.a, the NRC requested justification for the use of the screening criteria from that standard. In response to RAI 10.a, the licensee stated that use of the ASME/ANS RA-Sb-2013 PRA standard for screening external events meets all the requirements in the ASME/ANS RA-Sa-2009 PRA (Reference 10) standard because the differences between the screening criteria in the two versions of the PRA standard are primarily minor editorial changes. The licensee also presented a comparison of the fundamental criteria from the 2009 and 2013 versions of the standard demonstrating this conclusion. The licensee stated that the one important exception to this is that Progressive Screening criterion, "PS1," did not transition from the 2009 to the 2013 version of the PRA standard. However, the licensee explained that no hazards in the Hatch LAR were screened using PS1. The licensee stated that the set of screened hazards for Hatch are the same using either version of the standard. The NRC staff's review of the differences between the screening criteria in the 2009 and 2013 versions of the PRA Standard finds that the differences do not change the substance of the criteria. Also, the NRC staff noted that PS1, which was the only substantive change between the 2009 to 2013 versions, was not used by the licensee for this application because it would not have applied. Therefore, the NRC staff finds that the licensee's use of the criteria in the 2013 version of the PRA Standard for non-seismic external hazards and other hazards is an acceptable plant-specific alternative to the NRC-endorsed approach to support this application. This SE does not provide approval of the ASME/ANS 2013 Standard beyond the plant-specific applicability for Hatch.

In RAI 10.b, the NRC staff requested identification of the external hazards that would be evaluated using the flow chart in NEI 00-04, Section 5.4, Figure 5-6 as opposed to treating all SSCs as LSS with respect to non-seismic external hazards and other hazards. In response to RAI 10.b, the licensee explained that all other external hazards will be evaluated according to the flow chart in NEI 00-04, Section 5.4, Figure 5-6 and provided a list of hazards. They are hazards from Part 6 of the PRA standard (except internal flooding, internal fire, and seismic events which are addressed using PRAs). In LAR Attachment 4, the licensee provided justification for the screening criteria chosen for these hazards. The licensee's response to RAI 10.e also confirmed that all "other external hazards" including extreme winds and tornado hazards were screened out. In response to RAI 10.d, the licensee identified the type of SSCs that would be credited in screened scenarios. These SSCs included Seismic Category 1 Structures and other engineered features for defense against extreme wind and missile

protection; doors for defense against external flooding; and components associated with lightning protection. The license also confirmed that all SSCs credited for screening of the external hazards would be evaluated using the NEI 00-04, Section 5.4, Figure 5-6 guidance. The licensee stated that if the SSC was credited in the screening of a hazard then the impact of its removal from the screening will be considered by the IDP in its determination of the safety significance designation of the SSC.

Concerning external flooding, the licensee explained in response to RAI 10.b that this hazard was screened using the C1 criterion, meaning "[t]he hazards is of equal or lesser damage potential than the hazards for which the plant has been designed." The licensee stated that use of the C1 criterion is based on the results of calculations and analyses for the Hatch Flood Hazard Reevaluation Report (Reference 39), (Reference 40) (Non-Public), (Reference 41) (Non-Public), (Reference 42) and (Reference 43) (Non-Public). These analyses included examination of external flooding mechanisms:

- Local Intense Precipitation;
- Combined effects of the Probable Maximum Flood (PMF) with upstream overtopping dam failure with wind-induced waves;
- Flooding in rivers and streams (all season PMF with a half-PMF antecedent storm);
- Seismic upstream dam failure; and
- PMF with upstream overtopping dam failure).

The licensee explained in its response to RAI 10.b the calculated flood height for each mechanism. As documented in the NRC letter dated April 13, 2017 (Reference 43), there is a section on Local Intense Precipitation (LIP). The NRC staff noted that the flood height reached, as a result of these flood mechanisms, is either bounded by the lowest floor of the intake structure or the adverse impacts from the flooding such as water ingress would not cause a plant transient or impact mitigating systems safety functions due to site topography and plant design. In addition, the licensee stated that only passive features were credited in the screening of the external flooding hazard and that personnel actions were not relied upon to respond to the external flooding mechanisms. The NRC staff also noted that the licensee's disposition of these flooding mechanisms does not credit any FLEX strategies. The NRC staff's review finds the that the licensee has provided sufficient justification for the consideration of external flooding in the categorization of SSC and that the licensee's SSC categorization process will evaluate the safety significance of SSCs for the external flooding hazard consistent with the guidance provided in NEI 00-04, as endorsed by the NRC.

Concerning extreme wind and tornadoes, the licensee explained in response to RAI 10.b and RAI 10.c that this hazard was also screened using the C1 criterion, meaning "[t]he hazard is of equal or lesser damage potential than the hazards for which the plant has been designed." While the licensee's response used inconsistent language to describe the C1 criterion in its responses, the NRC staff noted that the C1 criterion has the same definition for all hazards evaluated according to the flow chart in NEI 00-04, Section 5.4. According to the licensee this is a change from the screening criterion identified in the LAR, which had previously provided that "design basis mean event frequency is < 1×10^{-5} /yr and the mean conditional core damage probability damage is <0.1" (i.e., criterion "PS3").

The licensee explained that based on the data provided in NUREG/CR-4461, Table 6-1 "Tornado Wind Speed Estimates for United States Nuclear Power Plants Sites" (Reference 44), the wind speed for Hatch associated with a tornado strike probability of 1×10^{-6} is 181 miles per

hour (mph) and 213 mph for a tornado strike probability of $1x10^{-7}$, using the Enhanced Fujita Scale. The licensee further explained that per the Hatch Updated Final Safety Analysis Report (UFSAR) Section 3.3, all Seismic Category 1 structures are designed for a tornado loading based on a 300 mph wind speed. According to the licensee, Non-Category 1 structures are designed to comply with Seismic II/I (two-over-one) requirements. The NRC staff noted that in UFSAR Section 3.3.2, failure of Seismic Category II structures not designed for tornado loads would not be affect the ability of Seismic Category I structures to perform their functions. As indicated in UFSAR Section 3.3.2, the licensee stated that safety-related systems and components necessary for safe shutdown are located within Seismic Category 1 structures or otherwise have been evaluated in engineering calculations as being able to withstand the design hazard.

The license also stated that in response to Regulatory Issue Summary (RIS) 2015-06 (Reference 45), a Tornado Missile Vulnerability Evaluation was performed. The evaluation identified two non-conformances that were subsequently resolved. One non-conformance was resolved by adding physical protection and the other was resolved through engineering calculations showing non-vulnerability due to inherent robustness.

The licensee explained that the frequency of a tornado that exceeds the plant design basis is less than $1x10^{-7}$ per year and tornado wind non-conformances with the design basis associated with protection of SSCs from missiles using other engineered features have been resolved as stated in Attachment 4 of the LAR. The NRC staff reviewed the above information and supporting documentation and finds that the licensee reviewed of the design basis and confirmed that plant SSCs credited for high winds and tornado protection will be evaluated in accordance with the guidance of Figure 5-6 of NEI 00-04 to ensure that no unscreened scenarios are created. Therefore, the NRC staff finds the licensee's SSC categorization process will evaluate the safety significance of SSCs for the high winds and tornados hazard consistent with the guidance provided in NEI 00-04, as endorsed by the NRC.

In summary, the licensee confirmed that the categorization process will not deviate from the guidance presented in NEI 00-04 for the evaluation of non-seismic external hazards and other hazards. The licensee clarified that as part of the categorization process, an assessment of all hazards except internal flood, internal fire, and seismic events would be performed to determine if there are components being categorized that participate in screened scenarios and whose failure would result in an unscreened scenario, consistent with the flow chart in Figure 5-6 in Section 5.4 of NEI 00-04.

The NRC staff's review finds that the licensee's categorization process will evaluate the safety significance of SSCs for non-seismic external hazards consistent with the guidance provided in Figure 5-6 of NEI 00-04, Revision 0, as endorsed by the NRC in RG 1.201, Revision 1.

3.2.4.4.2 Shutdown Risk

Consistent with the guidance in NEI 00-04, Revision 0, the licensee proposed using the shutdown safety assessment based on NUMARC 91-06 (Reference 15). NUMARC 91-06 provides considerations for maintaining defense-in-depth (DID) for the five key safety functions during shutdown, namely, decay heat removal capability, inventory control, power availability, reactivity control, and containment - primary/secondary. NUMARC 91-06 also 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 LAR Section 3.2.5, the licensee stated that consistent with the guidance in NEI 00-04, Section 5.5, only SSCs that meet either of the two following criteria will be considered preliminary HSS: (1) if it is considered to be part of a "primary safety system," or (2) its failure would initiate an event during shutdown plant conditions (e.g., loss of shutdown cooling).

- With regard to the first criterion, NEI 00-04, Section 5.5 states that when multiple systems/trains are available to satisfy the key safety function, only SSCs that support the primary and first alternative methods to satisfy the key safety function are considered to be the "primary shutdown safety systems."

- The SSC's failure would initiate an event during shutdown plant conditions (e.g., loss of shutdown cooling).

The use of NUMARC 91-06 described by the licensee in the submittal is consistent with the guidance in NEI 00-04, Revision 0, as endorsed by the NRC in RG 1.201, Revision 1. The approach uses an integrated and systematic process to identify HSS components, consistent with the shutdown evaluation process. Therefore, the NRC staff finds that the licensee's use of NUMARC 91-06 is acceptable, and is sufficient to demonstrate meeting the requirements set forth in 10 CFR 50.69(c)(1)(ii).

3.2.4.4.3 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 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, Revision 0, or RG 1.201, Revision 1, for passive component categorization, but was approved by the NRC for ANO-2 (Reference 16). The ANO-2 plant-specific approval considered a risk-informed methodology for safety classification and treatment program for repair/replacement activities for Class 2 and Class 3 pressure-retaining items or their associated supports (exclusive of Class CC and MC items), using a modification of the ASME Code Case N-660, "Risk-Informed Safety Classification for Use in Risk-Informed Repair/Replacement Activities, Section XI, Division 1" (Reference 46). The ANO-2 approval relied 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 because this approach will not allow categorization to be affected by changes in frequency arising from changes to the treatment. For ANO, the NRC staff found that the use of the repair/replacement methodology was acceptable for passive component categorization of Class 2 and Class 3 SSCs. The NRC did not approve the ANO methodology for generic use.

In Section 3.1.2 of the LAR the licensee submitted a plant-specific passive categorization process for Hatch that is intended to apply the same risk-informed process authorized in the

ANO-2 relief request alternative for the passive categorization of Class 2, 3, and non-class components. The licensee stated that "[a]ll ASME Code Class 1 SSCs with a pressure retaining function, as well as supports, will be assigned HSS, for passive categorization, which will result in HSS for its risk-informed safety classification and cannot be changed by the IDP." The NRC staff finds the licensee's proposed plant-specific request for passive categorization acceptable for use in the 10 CFR 50.69 categorization process.

3.2.4.5 Risk Sensitivity Study

Paragraph 50.69(c)(1)(iv) of 10 CFR requires, in part, that evaluations provide reasonable confidence that SSCs categorized as RISC-3, sufficient safety margins are maintained and that any potential increases in CDF and LERF resulting from changes to treatment are small. The categorization process described in NEI 00-04, Revision 0, includes an overall risk sensitivity study for all the LSS components to assure that if the unreliability of the components is increased, the increase in risk would be small (i.e., meet the acceptance guidelines of RG 1.174, Revision 3).

Section 3.1.1 of the LAR stated that an unreliability factor of 3 will be used for the sensitivity studies described in Section 8, "Risk Sensitivity Study," of NEI 00-04, Revision 0. Additionally, Section 3.2.7 of the LAR further confirmed that a cumulative sensitivity study will be performed where the failure probabilities (unreliability and unavailability, as appropriate) of all LSS components modeled in PRAs for all systems that have been categorized are increased by a factor of 3. The NRC staff finds the application of a factor of 3 for the sensitivities is consistent with the guidance in NEI 00-04, Revision 0, as endorsed by RG 1.201, Revision 1.

In Section 3.1.1 of the LAR, for the "Overall Categorization Process," the licensee quoted RG 1.201 as saying "'the implementation of all processes described in NEI 00-04 (i.e., Sections 2 through 12) is integral to providing reasonable confidence' and that 'all aspects of NEI 00-04 must be followed to achieve reasonable confidence in the evaluations required by 10 CFR 50.69(c)(1)(iv).'" This sensitivity study, together with the periodic review process discussed in Section 3.2.5 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 Section 8 of NEI 00-04, Revision 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).

3.2.4.6 Integrated Decision Making

RG 1.174 provides guidance to ensure all safety impacts of proposed licensing basis changes are evaluated in an integrated manner. The integrated evaluation ensures that a single key principle is not assessed in isolation, but in a manner that complements the NRC's deterministic approach and supports the NRCs traditional defense-in-depth philosophy.

Appendix B of SRP Chapter 19, Section 19.2 provides guidance and the staff expectations for the licensee's integrated decision-making process. The Appendix states in part, "[r]isk-informed applications are expected to require a process to integrate traditional engineering and probabilistic considerations to form the basis for acceptance." NEI 00-04 identifies two steps in

the categorization process that are responsible for the integrated assessment of the traditional engineering analyses and the risk results from the PRA and non-PRA assessments: (1) Preliminary Engineering Categorization of Function and (2) IDP Review and Approval. The NRC staff review of the two steps to ensure the processes is well-defined, systematic, repeatable, and scrutable are provided as follows:

Preliminary Engineering Categorization of Functions

All the information collected and evaluated in the licensee's engineering evaluations is provided to the IDP as described in Section 7 of NEI 00-04, Revision 0. The IDP will make the final decision about the safety significance of SSCs based on guidelines in NEI 00-04, Revision 0, the information it receives, and its expertise.

In Section 3.1.1 of the LAR the licensee stated, in part, "if any SSC is identified as HSS from either the integrated PRA component safety significance assessment (Section 5 of NEI 00-04) or the defense-in-depth assessment (Section 6), the associated system function(s) would be identified as HSS." The licensee also stated that, "[o]nce a system function is identified as HSS, then all the components that support that function are preliminary HSS."

The NRC staff finds that the above description provided by the licensee for the preliminary categorization of functions is consistent with NEI 00-04, Revision 0, as endorsed in RG 1.201, Revision 1, and is therefore, acceptable to meet the requirements set forth in 10 CFR 50.69(c)(1)(ii).

IDP Review and Approval

As required by 10 CFR 50.69(c)(2), 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. In Section 3.1.1 of the LAR, the licensee stated 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 IDP will comprise the required expertise.

The guidance in NEI 00-04, Revision 0, as endorsed in RG 1.201, Revision 1, 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). In Section 3.1.1 of the LAR, the licensee discusses that at least three members of the IDP will have a minimum of five years of experience at the plant, and there will be at least one member of the IDP who has a minimum of three years of experience in modeling and updating of the plant specific PRA. The licensee further stated that the IDP will be trained in the specific technical aspects and requirements related to the categorization process. This training, according to the licensee, 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 D-I-D 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, Revision 0, as endorsed in RG 1.201, Revision 1.

The IDP may change the categorization of a component from preliminary LSS to HSS based on its assessment and decision-making. As detailed in Section 3.1.1 of the LAR, the IDP's authority to change component categorization from preliminary HSS to LSS is however limited and only available to the IDP based upon the prescribed steps in the NEI 00-04 guidance as endorsed by RG 1.201, Revision 1. Consistent with the guidance in NEI 00-04, Revision 0, LAR Table 3-1 shows that components found to be preliminary HSS from the following aspects of the process cannot be re-categorized by the IDP: IEPRA, integrated PRA importance measures, shutdown, passive categorization, and DID. Further, LAR Table 3-1 shows SSCs identified as preliminary HSS through an SPRA, FPRA, or through the sensitivity studies outlined in Section 5 of NEI 00-04, may be presented to the IDP for categorization as LSS. According to the guidance, this can be done if this determination is supported by the integrated assessment process and other elements of the categorization process.

Section 9.2.2, "Review of Safety Related Low Safety-Significant Functions/SSCs," of NEI 00-04, Revision 0, states the IDP should determine whether functions/SSCs that have been identified as candidate LSS "are not implicitly depended upon to maintain safe shutdown capability, prevention of core damage and maintenance of containment integrity. In making their assessment, the IDP should consider the impact of loss of the function/SSC against the remaining capability to perform the basic safety functions . . . ." This section also provides seven specific questions that should be considered by the IDP for making the final determination of the safety-significance for each function/SSC.

The licensee confirmed in the notes to LAR Table 3-1 that the final assessment of the seven qualitative questions in Section 9.2 of NEI 00-04 (cited above) is the IDP's responsibility and that the final categorization of the function will be HSS when any one of the seven questions cannot be confirmed (false response) for that function.

The NRC staff finds the licensee's description of IDP acceptable and consistent with the guidance in NEI 00-04, Revision 0, as endorsed in RG 1.201.

3.2.4.7 Key Principle 4 Conclusion

The above sections summarized how the licensee's SSC categorization process is consistent with the guidance and methodology prescribed in NEI 00-04, Revision 0, and RG 1.201, Revision 1. As it is consistent with NEI 00-04, Revision 0, and RG 1.201, Revision 1 and the regulatory requirements in 10 CFR 50.69 it satisfies the fourth key principle for risk-informed decision-making prescribed in RG 1.174, Revision 3.

3.2.5 Key Principle 5: Monitor the Impact of the Proposed Change

RG 1.174 also establishes the need for an implementation and monitoring program to ensure that LB changes do not degrade operational safety over time and that no adverse effects occur from unanticipated degradation or common-cause mechanisms. The purpose of an implementation and monitoring program is to ensure that the impact of the proposed change continues to reflect the reliability and availability of the evaluated SSCs.

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, Revision 0 provides guidance that includes programmatic configuration control and a periodic review to ensure that the all aspects of the 10 CFR 50.69 program (i.e., includes traditional engineering analyses) and PRA models used to perform the risk assessment continue to reflect the as-built-as-operated plant and that plant modifications and updates to the PRA over time are continually incorporated.

Section 11 of NEI 00-04, Revision 0, as endorsed in RG 1.201, provides guidance on program documentation and change control while Section 12 discusses periodic review. These sections are described in NEI 00-04 with respect to satisfying 10 CFR 50.69(f) and 10 CFR 50.69(e), respectively. A more detailed staff review is provided below.

3.2.5.1 Periodic Review

Section 50.69(e) of 10 CFR, "Feedback and Process Adjustment," requires periodic updates to the licensee's PRA and SSC categorization. The NRC staff finds that changes over time to the PRA and to the SSC reliabilities are inevitable, and such changes are recognized by the 10 CFR 50.69(e) requirement for periodic updates.

Section 11.2 of NEI 00-04, Revision 0, titled, "Following Initial Implementation," states in part, "[t]he periodic update of the plant PRA may affect the results of the categorization process. If the results are affected, the licensee must make adjustments as necessary to either the categorization or treatment processes to maintain the validity of the processes." In Section 3.2.6 of the LAR, the licensee described the risk management process for maintaining and updating the Hatch PRA models used for the 10 CFR 50.69 categorization process. The described process provides provisions to ensure that the PRA models used to support the categorization reflect the as-built, as-operated plant. The licensee's process includes provisions for: monitoring potential areas affecting the PRA models (e.g., due to changes in the plant, errors or limitations identified in the model, industry operational experience); assessing the risk impact of unincorporated changes; and controlling the model and associated computer files. The process also includes reevaluating previously categorized systems to ensure the continued validity of the categorization. Routine PRA updates are performed every two refueling cycles at a minimum.

The NRC staff finds the risk management process as described by the licensee in the LAR consistent with Section 12 of NEI 00-04, Revision 0, as endorsed by the NRC in RG 1.201, Revision 1. The NRC staff also finds that this description is consistent with the requirements for feedback and process adjustments required by 10 CFR 50.69(e), and is, therefore, acceptable.

3.2.5.2 Program Documentation and Change Control

Section 50.69(f) of 10 CFR requires, in part, program documentation, change control, and records. In Section 3.2.6 of the LAR, the licensee stated that it will implement a process that addresses the requirements in Section 11 of NEI 00-04, Revision 0, pertaining to program documentation and change control records. Section 3.1.1 of the LAR stated that the SSC 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

Attachment 1 of the LAR states the following steps/elements are included in the licensee's 10 CFR 50.69 programmatic procedures: (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; (6) overall risk sensitivity study; (7) periodic review; and (8) documentation requirements identified in Section 3.1.1 of the LAR. Additional actions (e.g., final procedures and proposed alternative treatment) need not, and have not, been developed, submitted, or reviewed by the staff for issuance of the SE, but will be completed before implementation of the program.

The NRC staff recognizes that for facilities licensed under 10 CFR Part 50, Appendix B Criterion VI, for Document Control, procedures are considered formal plant documents requiring that, "[m]easures shall be established to control the issuance of documents, such as instructions, procedures, and drawings, including changes thereto, which prescribe all activities affecting quality." The NRC staff finds that the elements provided in Section 3.1.1 of the LAR, in addition to the list of prerequisites provided in Attachment 1 of the LAR for the Hatch 10 CFR 50.69 categorization process will be documented in formal licensee procedures, consistent with Section 11 of NEI 00-04, Revision 0, as endorsed by the NRC in RG 1.201, Revision 1, and therefore, are consistent with the requirements in 10 CFR 50.69(f)for program documentation, change control and records.

3.2.5.3 Conclusion for Principle 5

Based on its review of the LAR, the NRC staff finds that the change control and performance monitoring of categorized SSCs and PRA updates will sufficiently monitor the component performance to ensure that significant increases in failure rates of categorized components are detected and addressed. In addition, the PRA update program and associated reevaluation of component importance will appropriately consider the effects of changing equipment failure rates and changing plant configuration on the component safety-significant categories. As discussed above, the NRC staff finds the process in NEI 00-04 and the LAR will meet the requirements of 10 CFR 50.69(e) and 10 CFR 50.69(f), respectively. Therefore, the process used to characterize the SSC importance will reasonably reflect the current plant configuration and operating practices and applicable plant and industry operational experience as required in 10 CFR 50.69(c)(1)(ii). It also thus satisfies the fifth key principle for risk-informed decision-making prescribed in RG 1.174, Revision 3.

3.3 Technical Conclusion

The NRC staff finds the PRAs and the use of non-PRA methods along with the deterministic considerations outlined in NEI 00-04 for use in the SSC categorization process described by the licensee in the submittal, as supplemented in letters dated July 16, and December 18, 2019,

and are acceptable for use in the SSC categorization process. The NRC staff approves the use of the following approaches and methods in the licensee's 10 CFR 50.69 categorization process:

- IEPRA and IFPRA to assess internal events and internal flood risk, respectively
- FPRA to assess internal fire risk
- SPRA to assess seismic event risk
- Screening using criteria in Part 6 of the ASME/ANS RA-Sb-2103 PRA standard for non-seismic external hazards (e.g., high winds, external floods) and other hazards
- Shutdown Safety Management Plan consistent with NUMARC 91-06 to assess shutdown risk
- Plant-specific Hatch adoption of ANO-2 passive categorization method to assess passive components risk for Class 2 and 3 SSCs and their associated supports

The NRC staff also notes that the resolution of issues discussed in RAI 03 and as part of the resolution of issues discussed in RAI 12 involves a legal obligation added to the RFOL Condition, which is presented in Section 4.0 of this SE.

The legal obligation ensures that the FPRA that will be used in 10 CFR 50.69 categorization will reflect the NFPA-805 plant modifications and implementation items that are completed and that the RG 1.174, Revision 3 risk acceptance guidelines (i.e., total CDF less than or equal to 1×10^{-5} and total LERF less than or equal to 1.0×10^{-6}) are confirmed to be met prior to implementation of the 10 CFR 50.69 categorization process.

The NRC staff reviewed all of the primary steps outlined in Section 3.2.1 of this SE used by the licensee in the 10 CFR 50.69 categorization process to assess the safety significance of active and passive components while ensuring the SSC's intended functions remain intact. Based on its review, the NRC staff concludes that the licensee's categorization process adequately implements 10 CFR 50.69 using models, methods, and approaches consistent with NEI 00-04, Revision 0, and RG 1.201, Revision 1, and therefore, satisfies the requirements of 10 CFR 50.69(c). Based on its review, the NRC staff finds the licensee's proposed categorization process, conditioned on the proposed license condition, discussed in Section 4.0 of this SE, is acceptable for categorizing the safety significance of SSCs. Specifically, the NRC staff concludes that the licensee's categorization process:

(1) Considers results and insights from plant-specific IEPRA, IFPRA, FPRA, and SPRA that have been subjected to a peer review process against RG 1.200, Revision 2, as reviewed in Sections 3.2.4.3 of this SE, and are of sufficient quality and level of detail to support the categorization process, and therefore, meets the requirements in 10 CFR 50.69(c)(1)(i);

(2) Determines SSC functional importance using an integrated, systematic process that reasonably reflects the current plant configuration, operating practices, and applicable plant and industry operational experience, as reviewed in Section 3.2 of this SE and therefore, meets the requirements in 10 CFR 50.69(c)(1)(ii);

(3) Maintains DID, as reviewed in Section 3.2.2 of this SE, and therefore, meets the requirements in 10 CFR 50.69(c)(1)(iii);

(4) Includes evaluations that provide reasonable confidence that for SSCs categorized as RISC-3, sufficient safety margins are maintained and that any potential increases in CDF and LERF resulting from changes in treatment are small, as reviewed in Section 3.2.3 and 3.2.4.5 of this SE, and therefore, meets the requirements in 10 CFR 50.69(c)(1)(iv);

(5) Is performed for entire systems and structures, rather than for selected components within a system or structure, as reviewed in Section 3.2.1 of this SE, and therefore, the requirements in 10 CFR 50.69(c)(1)(v) will be met upon implementation; and

(6) Includes categorization by IDP, staffed with expert, plant-knowledgeable members whose expertise includes, at a minimum, PRA, safety analysis, plant operation, design engineering and system engineering, as reviewed in Section 3.2.4.6 of this SE, and therefore, meets the requirements in 10 CFR 50.69(c)(2).

In Attachment 1 of the LAR, the licensee stated that SNC has already established procedures implementing the categorization process on plant systems. The license provided a list of eight essential elements that are contained in these procedures which reflect the elements described in the NEI 00-04, Revision 0 and which are listed in Section 3.2.5.2 of this SE. The NRC staff's review of these elements as described by the licensee for implementation of the SSC categorization process is provided in Section 3.2 of this SE. Therefore, the NRC staff concludes that the Hatch SSC categorization process will be controlled consistent with 10 CFR 50.69(e) and (f) for ensuring that the categorization of SSCs continue to reflect the as-built, as-operated plant design.

4.0 CHANGES TO THE OPERATING LICENSE

Paragraph 50.69(b)(2) of 10 CFR requires the licensee to submit an application that describes the categorization process. Paragraph 50.69(b)(3) of 10 CFR states that the Commission will approve the licensee's implementation of this section if it determines that the categorization process satisfies the requirements of 10 CFR 50.69(c). As described in this SE, the NRC staff has concluded that the 10 CFR 50.69 categorization process described in the licensee's application, as supplemented, includes a description of the categorization process that satisfies the requirements of 10 CFR 50.69(c).

The NRC staff's finding on the acceptability of the PRA evaluation in the licensee's proposed 10 CFR 50.69 process is conditioned upon the License Condition provided below:

[SNC] is approved to implement 10 CFR 50.69 using the processes for categorization of RISC-1, RISC-2, RISC-3, and RISC-4 [SSCs] specified in the renewed license amendment dated June 26, 2020.

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

Prior to implementation of renewed license amendment dated June 26, 2020, SNC shall update the PRA models to reflect the as-built, as-operated, and as-maintained plant and shall ensure the risk acceptance guidelines found in RG 1.174, Rev. 3 are met.

Based on its review of the licensee's LAR as supplemented and the evaluation in this SE, the NRC staff finds that the proposed license condition is acceptable because it adequately implements 10 CFR 50.69 using models, methods, and approaches consistent with the

applicable guidance that has previously been endorsed by the NRC. 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 with regard to adequate completion of the implementation item, will be tracked and dispositioned appropriately in accordance with the requirements of 10 CFR 50.69(f) and 10 CFR Part 50, Appendix B Criterion VI, and could be subject to NRC enforcement action(s).

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Georgia State official was notified of the proposed issuance of the amendments on March 16, 2020. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of facility components 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 that the amendments involve no significant hazards consideration, and there has been no public comment on such finding published in the *Federal Registe*r (84 FR 11340 dated March 26, 2019). 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

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

8.0 REFERENCES

1 Gayheart, C. A., Southern Nuclear Operating Company (SNC) letter to U. S. Nuclear Regulatory Commission (NRC), "Edwin I. Hatch Nuclear Plant, Units 1 and 2, Application to Adopt 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors," June 7, 2018 (ADAMS Accession No. ML18158A583).

2 Gayheart, C.A., Southern Nuclear Operating Company (SNC), letter to U. S. Nuclear Regulatory Commission (NRC), "Edwin I. Hatch Nuclear Plant, Units 1 and 2, Response to Request for Additional Information Regarding Application to Adopt Title 10 of the Code of Federal Regulations (CFR) 50.69," July 16, 2019 (ADAMS Accession No. ML19197A097).

3 Gayheart, C. A., Southern Nuclear Operating Company (SNC), letter to U. S. Nuclear Regulatory Commission (NRC), "Edwin I. Hatch Nuclear Plant, Units 1 and

2, Response to Request for Additional Information Regarding Application to Adopt Title 10 of the Code of Federal Regulations (CFR) 50.69," Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors," December 18, 2019 (ADAMS Accession No. ML19352F705).

4 Gayheart, C. A., Southern Nuclear Operating Company (SNC), letter to U. S. Nuclear Regulatory Commission (NRC), "Edwin I. Hatch Nuclear Plant, Units 1 and 2, Response to Request for Additional Information (RAI) Regarding Application to Adopt 10 CFR 50.69," "Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors," June 2, 2020 (ADAMS Accession No. ML20154K716).

5 U. S. Nuclear Regulatory Commission (NRC), letter to C. A. Gayheart, Southern Nuclear Operating Company (SNC), "Edwin I. Hatch Nuclear Plant, Units 1 and 2, Regulatory Audit forthe Review of License Amendment Requests to Adopt the Risk-Informed Provisions of Title 10 of the Code of Federal Regulations," Paragraph 50.48(c) and 50.69," January 23, 2019 (ADAMS Accession No. ML19018A231).

6 Lamb, John, U. S. Nuclear Regulatory Commission (NRC), EMail to Joyce Ryan, Southern Nuclear Operatiing Company (SNC), "Request for Additional Information - Edwin I. Hatch Nuclear Plant, Units 1 and 2, Application to Adopt Title 10 of the Code of Federal Regulations 50.69," Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors," April 17, 2019 (ADAMS Accession No. ML19107A496).

7 U. S. Nuclear Regulatory Commission (NRC), "Regulatory Guide (RG) 1.201, Revision 1 - Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance," May 2001 (ADAMS Accession No. ML061090627).

8 Nuclear Energy Institute (NEI), "NEI 00-04, Revision 0, 10 CFR 50.69 SSC Categorization Guideline," July 2005 (ADAMS Accession No. ML052900163).

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

10 American Society of Mechanical Engineers (ASME) and American Nuclear Society (ANS), "Addendum A to RA-S-2008, PRA Standard ASME/ANS RA-Sa-2009, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," June 2005.

11 U. S. Nuclear Regulatory Commission (NRC), "Regulatory Guide (RG) 1.174, Revision 3, An Approach for Using Probabilistic Risk Assessment [PRA] in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," January 2018 (ADAMS Accession No. ML17317A256).

12 U. S. Nuclear Regulatory Commission (NRC), "NUREG-1855, Revision 1, Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking. Final Report," March 2017 (ADAMS Accession No. ML17062A466).

13 U. S. Nuclear Regulatory Commission (NRC), "NUREG-0800, Standard Review

Plan, Chapter 19, Section 19.2 - Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance," June 2007 (ADAMS Accession No. ML071700658).

14 U. S. Nuclear Regulatory Commission (NRC), "NUREG-0800, Standard Review Plan, Chapter 19, Section 19.1 - Determining the Technical Adequacy of Probabilistic Risk Assessment for Risk-Informed License Amendment Requests After Initial Fuel Load," September 2012 (ADAMS Accession No. ML12193A107).

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

16 Markley, Michael T., U. S. Nuclear Regulatory Commission (NRC), letter to Arkansas Nuclear One, Entergy Operations, Inc., "Arkansas Nuclear One, Unit 2, Approval of Request for Alternative ANO2-R&R-004, Revision 1, Request to Use Risk-Informed Safety Classification and Treatment for Repair/Replacement Activities in Class 2 and 3 Moderate and High Energy Systems," April 22, 2009 (ADAMS Accession No. ML090930246).

17 Gayheart, C. A., Southern Nuclear Operating Company (SNC), letter to U. S. Nuclear Regulatory Commission (NRC), "Edwin I. Hatch Nuclear Plant, Units 1 and 2, License Amendment Request to Adopt NFPA-805 Performance Based Standard for Fire Protection for LIght Water Reactor Electric Generating Plants (2001 Edition)," April 4, 2018 (ADAMS Accession No. ML18096A936).

18 U. S. Nuclear Regulatory Commission (NRC), "NRC RIS 2007-06 "NRC Regulatory Issue Summary 2007-06 Regulatory Guide 1.200 Implementation," March 22, 2007 (ADAMS Accession No. ML070650428).

19 General Electric Hitachi; Westinghouse Electric Company; and AREVA, NP, Inc. - Prepared for Nuclear Energy Institute (NEI), "NEI 05-04 - Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard," November 2008 (ADAMS Accession No. ML083430462).

20 Anderson, Victoria K., Nuclear Energy Institute (NEI), letter to U. S. Nuclear Regulatory Commission (NRC), "Final Revision of Appendix X to NEI 05-04/07-12/12-16, Closeout of Facts and Observations," February 21, 2017 (ADAMS Accession No. ML17086A431).

21 Nuclear Energy Institute (NEI), "NEI 07-12, Revision 1 - Fire Probabilistic Risk Assessment Peer Review Process Guidelines," Washington, DC, June 2010 (ADAMS Accession No. ML102230070).

22 Nuclear Energy Institute (NEI), "NEI 12-13 - External Hazards PRA [Probabilistic Risk Assessment] Peer Review Process Guidelines," Washington, DC, August 2012 (ADAMS Accession No. ML12240A027).

23 U. S. Nuclear Regulatory Commission (NRC), letter to Nuclear Energy Institute (NEI), "U. S. Nuclear Regulatory Acceptance on Nuclear Energy Institute Appendix X to Guidance 05-04, 07-12, and 12-13, Close-Out of Facts and Observations (F&O's)," May 3, 2017 (ADAMS Accession No. ML17079A427).

24 Gayheart, C. A., Southern Nuclear Operating Company (SNC), letter to U. S. Nuclear Regulatory Commission (NRC), "Edwin I. Hatch Nuclear Plant, Units 1 and

2, Response to Request for Additional Information Regarding Application to Adopt Title 10 of the Code of Federal Regulations (CFR) 50.69," Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors," December 18, 2019 (ADAMS Accession No. ML19352F705).

25 U. S. Nuclear Regulatory Commission (NRC), letter to Southern Nuclear Operating Company (SNC), "Edwin I. Hatch Nuclear Plant, Units 1 and 2, Issuance of Amendments 304 and 249, Regarding License Amendment Request to Adopt NFPA-805 Performance Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," June 11, 2020 (ADAMS Accession No. ML20066F592).

26 Thomas, Brian E., U. S. Nuclear Regulatory Commission (NRC) letter to C. R. Grantom P. E. & Assoc. LLC and Robert J. Budnitz, Lawrence Berkeley National Laboratory, "U. S. Nuclear Regulatory Commission Acceptance of American Society of Mechanical Engineers (ASME) and American Nuclear Society (ANS) - ASME/ANS RA-S Case 1," Case for ASME/ANS RA-Sb-2013 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment of Nuclear Power Plant Applications," March 12, 2018 (ADAMS Accession No. ML18017A964).

27 Franovich, Mike, U. S. Nuclear Regulatory Commission (NRC), letter to Greg Krueger, Nuclear Energy Institute (NEI), "U. S. Nuclear Regulatory Commission Acceptance of Nuclear Energy Institute (NEI) Guidance NEI 12-13 - External Hazards PRA Peer Review Process Guidelines (August 2010)," March 7, 2018 (ADAMS Accession No. ML18025C024 and ML18025C025).

28 Correia, Richard, U. S. Nuclear Regulatory Commission (NRC), letter to O. Martinez, American Society of Mechanical Engineers (ASME), "U. S. Nuclear Regulatory Commission Comments on "Addenda to a Current ANS: ASME RA-SB-20XX, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," July 6, 2011 (ADAMS Accession No. ML111720067).

29 Hutto, J. J., Southern Nuclear Operating Company (SNC), letter to U. S. Nuclear Regulatory Commission (NRC), "Vogtle Electric Generating Plant, Units 1 and 2 - Response to Supplemental Information Needed for Acceptance of Systematic Risk-Informed Assessment of Debris Technical Report," July 17, 2017 (ADAMS Accession No. ML17192A245).

30 U. S. Nuclear Regulatory Commission (NRC), "NUREG-1855, Volume 1, - Guidance on the Treatment of Uncertainities Associated with PRAs in Risk-Informed Decision Making - Main Report," March 2009 (ADAMS Accession No. M090970525).

31 Electric Power Research Institute (EPRI), "TR-1026511 - Practical Guidance of the Use of Probabilistic Risk Assessment in Risk-Informed Applications with a Focus on the Treatment of Uncertainty," December 2012.

32 Electric Power Research Institute (EPRI), "TR-1016737 - Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments," December 2008.

33 U. S. Nuclear Regulatory Commission (NRC); Brookhaven National Laboratory; Sandia National Laboratories, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making - Main Report (Revision 0)," March 2009 (ADAMS Accession No. ML090970525).

34 Electric Power Research Institutre (EPRI), "NUREG/CR-6850, Final Report - EPRI/NRC-RES: Fire PRA Methodology for Nuclear Power Facilities (Volume 1: Summary & Overview)," September 2005.

35 Electric Power Research Institute (EPRI), "NUREG/CR-6850, Final Report - EPRI/NRC-RES: Fire PRA Methodology for Nuclear Power Facilities (Volume 2: Detailed Methodology)," September 2005.

36 Electric Power Research Institute (EPRI), "Fire Probabilistic Risk Assessment Methods Enhancements," September 2010.

37 Gayheart, C. A., Southern Nuclear Operating Company (SNC), letter to U. S. Nuclear Regulatory Commission (NRC), "Edwin I. Hatch Nuclear Plant, Units 1 and 2 - Response to Request for Additional Information Regarding the License Amendment Request to Transition to 10 CFR 50.48(c)," NFPA-805 Performance Based Standard for Fire Protection for LIght Water Reactor Generating Plants," December 13, 2019 (ADAMS Accession No. ML19351D130).

38 American Society of Mechanical Engineers (ASME) and American Nuclear Society (ANS), "ASME/ANS RA-Sb-2013 - Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications, Addendum B to RA-S-2008," 2013, ASME, New York, NY.

39 Pierce, C. R., Southern Nuclear Operating Company (SNC), letter to U. S. Nuclear Regulatory Commission (NRC), "Edwin I. Hatch Nuclear Plant, Units 1 and 2 - Recommendation 2.1 Flood Hazard Reevaluation Report - Requested by NRC Letter dated March 12, 2012," March 6, 2014 (ADAMS Accession No. ML14069A054).

40 Pierce, C. R., Southern Nuclear Operating Company (SNC), letter to U. S. Nuclear Regulatory Commission (NRC), "Edwin I. Hatch Nuclear Plant, Units 1 and 2 - Response to Request for Additional Information," August 6, 2014 (ADAMS Accession No. ML14219A570 - Non-Public).

41 Pierce, C. R., Southern Nuclear Operating Company (SNC), letter to U. S. Nuclear Regulatory Commission (NRC), "Edwin I. Hatch Nuclear Plant, Units 1 and 2 - Response to Supplemental Request for Additional Information, Recommendation 2.1 Flood Hazard Reevaluation Report," June 3, 2015 (ADAMS Accession No. ML15154B601 - Non-Public).

42 Pierce, C. R., Southern Nuclear Operating Company (SNC), letter to U. S. Nuclear Regulatory Commission (NRC), "Edwin I. Hatch Nuclear Plant, Units 1 and 2 - Response to Request for Docketing Information Need 1 and 8, Recommendation 2.1 Flood Hazard Reevaluation Report," March 07, 2016 (ADAMS Accession No. ML16069A087).

43 U. S. Nuclear Regulatory Commission (NRC), letter to J. J. Hutto, Southern Nuclear Operating Company, Inc. (SNC), "Edwin I. Hatch Nuclear Plant, Units 1 and 2 - Flood Hazard Mitigation Strategies Assessment," April 13, 2017 (ADAMS Accession No. ML17069A234 - Non-Public).

44 Pacific Northwest National Laboratory and U. S. Nuclear Regulatory Commission (NRC), "NUREG/CR-4461, Revision 2 - Tornado Climatology of the Contiguous United States," February 2007 (ADAMS Accession No. ML 070810400).

45 U. S. Nuclear Regulatory Commission (NRC), "NRC Regulatory Issue Summary 2015-06 - Tornado Missile Protection," June 10, 2015 (ADAMS Accession No. ML15020A419).

46 American Society of Mechanical Engineers (ASME), "ASME Code Case N-660 - Risk-Informed Safety Classification for Use in Risk-Informed Repair/Replacement Activities," July 2002.

47 Martin, Robert, U. S. Nuclear Regulatory Commission (NRC), letter to C. R. Pierce, Southern Nuclear Operating Company, Inc. (SNC), "Votle Electric Generating Plant, Units 1 and 2 - Issuance of Amendments (173 and 155) Re: Use of 10 CFR 50.69," December 17, 2014 (ADAMS Accession No. ML14237A034).

Principal Contributors: Mihaela Biro, NRR
Ching Ng, NRR

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DATE	03/16/2020	04/22/2020	03/13/2020	03/13/2020
OFFICE	OGC - NLO	NRR/LPL2-1/BC	NRR/LPL2-1/PM	
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