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

September 24, 2019

Mr. Ernest J. Kapopoulos, Jr.
Site Vice President
H. B. Robinson Steam Electric Plant
Duke Energy Progress, LLC
3581 West Entrance Road, RNPA01
Hartsville, SC 29550

SUBJECT: H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2 – ISSUANCE OF AMENDMENT NO. 266 TO ADOPT 10 CFR 50.69, "RISK-INFORMED CATEGORIZATION AND TREATMENT OF STRUCTURES, SYSTEMS, AND COMPONENTS FOR NUCLEAR POWER REACTORS" (EPID L-2018-LLA-0095)

Dear Mr. Kapopoulos:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 266 to Renewed Facility Operating License No. DPR-23 for the H. B. Robinson Steam Electric Plant, Unit No. 2 (Robinson). This amendment is in response to your license amendment request dated April 5, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18099A130), as supplemented by letters dated June 6, 2018 (ADAMS Accession No. ML18162A147), November 13, 2018 (ADAMS Accession No. ML18317A026), and May 6, 2019 (ADAMS Accession No. ML19126A143).

The amendment adds a new license condition to the renewed facility operating license to permit 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 subject to special treatment requirements (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation) based on a method of categorizing structures, systems and components according to their safety significance.

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

Sincerely,

Natreon J. Jordan, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-261

Enclosures:
1. Amendment No. 266 to DPR-23
2. Safety Evaluation

cc: Listserv

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

DUKE ENERGY PROGRESS, LLC

DOCKET NO. 50-261

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 266
Renewed License No. DPR-23

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

   A. The application for amendment by Duke Energy Progress, LLC (the licensee), dated April 5, 2018, as supplemented by letters dated June 6, 2018, November 13, 2018, and May 6, 2019, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the license as indicated in the Attachment to this license amendment, and paragraph 4 of Renewed Facility Operating License No. DPR-23 is hereby amended to read as follows:

   Additional Conditions

   The Additional Conditions contained in Appendix B, as revised through Amendment No. 266, are hereby incorporated into this license. Duke Energy Progress, LLC shall operate the facility in accordance with the additional conditions.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Undine Shoop, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility Operating License and Appendix B, "Additional Conditions"

Date of Issuance: September 24, 2019

ATTACHMENT TO LICENSE AMENDMENT NO. 266

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

RENEWED FACILITY OPERATING LICENSE NO. DPR-23

DOCKET NO. 50-261

Replace the following pages of Renewed Facility Operating License No. DPR-23 and Appendix B, Additional Conditions, with the attached revised pages. The revised pages are identified by amendment number and contains a marginal line indicating the area of change.

Renewed Facility Operating License

Remove Page	Insert Page
7	7

Appendix B, Additional Conditions

Remove Page	Insert Page
Appendix B-2	Appendix B-2
---------	Appendix B-3

4. <u>Additional Conditions</u>

The Additional Conditions contained in Appendix B, as revised through Amendment No. 266, are hereby incorporated into this license. Duke Energy Progress, LLC. shall operate the facility in accordance with the additional conditions.

5. This renewed license is effective as of the date of issuance and shall expire at midnight on July 31, 2030.

FOR THE NUCLEAR REGULATORY COMMISSION

ORIGINAL SIGNED BY
J. E. DYER

J. E. Dyer, Director
Office of Nuclear Reactor Regulation

Attachments: 1. Appendix A - Technical Specifications
2. Appendix B - Additional Conditions

Date of Issuance: April 19, 2004

	date of the most recent successful tracer gas test, or within the next 18 months if the time period since the most recent successful tracer gas test is greater than 6 years. (b) The first performance of the periodic assessment of CRE habitability, TS 5.5.17.c.(ii), shall be within the next 9 months. (c) The first performance of the periodic measurement of CRE pressure, TS 5.5.17.d, shall be within 18 months, plus the 138 days allowed by SR 3.0.2, as measured from the date of the most recent successful pressure measurement test.	
266	Duke Energy is approved to implement 10 CFR 50.69 using the processes for categorization of Risk Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 structures, systems, and components (SSCs) using: Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events, including internal flooding, internal fire, high winds, and external flood; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 SSCs and their associated supports; and the results of non PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards, i.e., seismic margin analysis (SMA) to evaluate seismic risk, and a screening of other external hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009; as specified in Unit 2 License Amendment No. 266 dated September 24, 2019.	Upon implementation of Amendment No. 266.

Duke Energy will complete the implementation items list in Attachment 1 of Duke letter to NRC dated May 6, 2019 prior to implementation of 10 CFR 50.69. All issues identified in the attachment will be addressed and any associated changes will be made, focused-scope peer reviews will be performed on changes that are PRA upgrades as defined in the PRA standard (ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, Revision 2), and any findings will be resolved and reflected in the PRA of record prior to implementation of the 10 CFR 50.69 categorization process.

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 266 TO

RENEWED FACILITY OPERATING LICENSE NO. DPR-23

DUKE ENERGY PROGRESS, LLC

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

DOCKET NO. 50-261

1.0 INTRODUCTION

By letter dated April 5, 2018 (Reference 1), as supplemented by letters dated June 6, 2018 (Reference 2), November 13, 2018 (Reference 3), and May 6, 2019 (Reference 4), Duke Energy Progress, LLC (Duke Energy, the licensee) submitted a license amendment request (LAR) for the H. B. Robinson Steam Electric Plant, Unit No. 2 (Robinson). The licensee proposed to add a new license condition to the renewed facility operating license to allow the implementation of Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.69, "Risk informed categorization and treatment of structures, systems and components [SSCs] for nuclear power reactors." The provisions of 10 CFR 50.69 allow adjustment of the scope of SSCs subject to special treatment requirements (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation) based on a method of categorizing SSCs according to their safety significance.

By e-mails dated October 12, 2018 (Reference 5), and March 12, 2019 (Reference 6), the U.S. Nuclear Regulatory Commission (NRC or the Commission) staff issued requests for additional information (RAIs) to the licensee. By letters dated November 13, 2018 (Reference 3), and May 6, 2019 (Reference 4), the licensee responded to the RAIs. The supplemental letters dated June 6, 2018, November 13, 2018, and May 6, 2019, 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* (FR) on June 5, 2018 (83 FR 26101).

2.0 REGULATORY EVALUATION

2.1 Risk-Informed Categorization and Treatment of SSCs

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

To take advantage of the safety enhancements available through the use of PRA, the NRC promulgated a new regulation, 10 CFR 50.69, in the *Federal Register* on November 22, 2004 (69 FR 68008), which became effective on December 22, 2004. The provisions of 10 CFR 50.69 allow adjustment of the scope of SSCs subject to special treatment requirements. Special treatment refers to those requirements that provide increased assurance beyond normal industry practices that SSCs perform their design-basis functions. For SSCs categorized as low-safety-significance (LSS), alternative treatment requirements may be implemented in accordance with the regulation.

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

Section 50.69 of 10 CFR does not allow for the elimination of SSC functional requirements or allow equipment that is required by the deterministic design basis to be removed from the facility. Instead, 10 CFR 50.69 enables licensees to focus their resources on SSCs that make a significant contribution to plant safety. In 2004, when promulgating the 10 CFR 50.69 rule, the Commission stated:

> It is important to note that this rulemaking effort, while intended to ensure that the scope of special treatment requirements imposed on SSCs is risk-informed, is not intended to allow for the elimination of SSC functional requirements or to allow equipment that is required by the deterministic design basis to be removed from the facility (i.e., changes to the design of the facility must continue to meet the current requirements governing design change; most notably § 50.59). Instead, this rulemaking should enable licensees and the staff to focus their resources on SSCs that make a significant contribution to plant safety by restructuring the regulations to allow an alternative risk-informed approach to special treatment. Conversely, for SSCs that do not significantly contribute to plant safety on an individual basis, this approach should allow an acceptable, though reduced, level of confidence (i.e., "reasonable confidence") that these SSCs will satisfy functional requirements. However, continued maintenance of the health and safety of the public will depend on effective implementation of § 50.69 by the licensee or applicant applying the rule at its nuclear power plant.

Final Rule, Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors, 69 Fed. Reg. 68008, 68011 (Nov. 2, 2004).

2.2 Licensee's Proposed Changes

The licensee proposed in the LAR to amend its renewed facility operating license. On October 12, 2018, the NRC staff requested further clarification of the license change in PRA RAI 09 (Reference 5). The licensee responded to PRA RAI-09 (Reference 3) by adding the following license condition that would allow for the implementation of 10 CFR 50.69 that was revised in the letter dated May 6, 2019 (Reference 4):

> Duke Energy is approved to implement 10 CFR 50.69 using the processes for categorization of Risk Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 structures, systems, and components (SSCs) using: Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events, including internal flooding, internal fire, high winds, and external flood; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 SSCs and their associated supports; and the results of non PRA evaluations that are based on the IPEEE [Individual Plant Examination of External Events] Screening Assessment for External Hazards, i.e., seismic margin analysis (SMA) to evaluate seismic risk, and a screening of other external hazards updated using the external hazard screening significance process identified in ASME/ANS [American Society of Mechanical Engineers/American Nuclear Society] PRA Standard RA-Sa-2009; as specified in Unit 2 License Amendment No. 266 dated September 24, 2019.
>
> Duke Energy will complete the implementation items list in Attachment 1 of Duke letter to NRC dated May 6, 2019 prior to implementation of 10 CFR 50.69. All issues identified in the attachment will be addressed and any associated changes will be made, focused-scope peer reviews will be performed on changes that are PRA upgrades as defined in the PRA standard (ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, Revision 2), and any findings will be resolved and reflected in the PRA of record prior to implementation of the 10 CFR 50.69 categorization process.
>
> Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from a seismic margins approach to a seismic probabilistic risk assessment approach).

2.3 Regulatory Review

The NRC staff reviewed the licensee's application to determine whether (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) activities proposed will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or the health and safety of the public. The staff considered the following regulatory requirements and guidance during its review of the proposed change.

Regulatory Requirements

Section 50.69 of 10 CFR provides an alternative approach for establishing requirements for treatment of SSCs for nuclear power reactors using a risk-informed method of categorizing SSCs according to their safety-significance. This regulation permits power reactor licensees to implement an alternative regulatory framework with respect to special treatment.

Section 50.69(a) of 10 CFR defines the following terms:

RISC-1 SSCs: Safety-related SSCs that perform safety significant functions
RISC-2 SSCs: Nonsafety-related SSCs that perform safety significant functions
RISC-3 SSCs: Safety-related SSCs that perform low safety significant functions
RISC-4 SSCs: Nonsafety-related SSCs that low safety significant functions

For these terms, "Safety significant function means a function whose degradation or loss could result in a significant adverse effect on defense-in-depth, safety margin, or risk."[1] 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).

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

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

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

(iii) Maintain Defense-In-Depth.

(iv) Include evaluations that provide reasonable confidence that for SSCs categorized as RISC–3, sufficient safety margins are maintained and that any potential increases in core damage frequency (CDF) and large early release frequency (LERF) resulting from changes in treatment permitted by implementation of § § 50.69(b)(1) and (d)(2) are small.

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[1] Nuclear Energy Institute (NEI) 00-04, Revision 0, "10 CFR 50.69 SSC Categorization Guideline (Reference 7), uses the term "high-safety-significant (HSS)" to refer to SSCs that perform safety significant functions. The NRC understands HSS to have the same meaning as "safety significant" (i.e., SSCs that are categorized as RISC-1 or RISC-2), as used in 10 CFR 50.69.

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

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

For RISC-1 and RISC-2 SSCs, 10 CFR 50.69 maintains current regulatory requirements (i.e., it does not remove any requirements from these SSCs) for special treatment. For RISC-3 SSCs, licensees can implement alternative treatment requirements in accordance with 10 CFR 50.69(b)(1) and 10 CFR 50.69(d). For RISC-4 SSCs, 10 CFR 50.69 does not impose new treatment requirements.

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

Guidance

Nuclear Energy Institute (NEI) 00-04, Revision 0, "10 CFR 50.69 SSC Categorization Guideline," dated July 2005 (Reference 7), describes a process for determining the safety-significance of SSCs and categorizing them into the four RISC categories defined in 10 CFR 50.69. This categorization process is an integrated decisionmaking process that incorporates risk and traditional engineering insights. The guidance in NEI 00-04, Revision 0, provides options for licensees implementing different approaches depending on the scope of their PRA models. It also allows the use of non-PRA approaches when PRAs have not been performed to address hazards such as seismic, fire, or shutdown risk. The NEI 00-04 guidance identifies non-PRA approaches such as fire-induced vulnerability evaluation to address fire risk, SMA to address seismic risk, and guidance in Nuclear Management and Resource Council (NUMARC) 91-06, "Guidelines for Industry Actions to Assess Shutdown Management" dated December 1991 (Reference 8), to address shutdown operations.

Sections 2 through 10 of NEI 00-04, Revision 0, describe a method for meeting the requirements of 10 CFR 50.69(c), as follows:

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

Additionally, Section 11 of NEI 00-04, Revision 0, provides guidance on program documentation and change control related to the requirements of 10 CFR 50.69(f), and 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).

NRC Regulatory Guide (RG) 1.201 (For Trial Use), Revision 1, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants according to their Safety Significance," dated May 2006 (Reference 9), endorses the categorization method described in NEI 00-04, Revision 0, with clarifications, limitations, and conditions. RG 1.201 states that the applicant is expected to document, at a minimum, the technical adequacy of the internal initiating events PRA. Licensees may use either PRAs or alternative approaches for hazards other than internal initiating events. The guidance in RG 1.201 clarifies that the NRC staff expects that licensees proposing to use non-PRA approaches in their categorization should provide a basis in the submittal for why the approach and the accompanying method employed to assign safety-significance to SSCs is technically adequate. The guidance further states that as part of the NRC's review and approval of a licensee's or applicant's application requesting to implement 10 CFR 50.69, the NRC staff intends to impose a license condition that will explicitly address the scope of the PRA and non-PRA methods used in the licensee's categorization approach. If a licensee or applicant wishes to change its categorization approach and the change is outside the bounds of the NRC's license condition (e.g., switch from a SMA to a seismic PRA), the licensee or applicant will need to seek NRC approval, via a license amendment, of the implementation of the new approach in its categorization process. The guidance in RG 1.201 also states that all aspects of NEI 00-04 must be followed to achieve reasonable confidence in the evaluations required by 10 CFR 50.69(c)(1)(iv).

RG 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," dated March 2009 (Reference 10), describes an acceptable approach for determining whether the quality of the PRA, in total or the parts that are used to support an application, is sufficient to provide confidence in the results, such that the PRA can be used in regulatory decisionmaking for light-water reactors. It endorses, with clarifications, the ASME/ANS PRA Standard ASME/ANS RA-Sa-2009 (hereafter known as the "ASME/ANS 2009 Standard" or "PRA Standard") (Reference 11). This RG provides guidance for determining the technical adequacy of a PRA by comparing the PRA to the relevant parts of the ASME/ANS 2009 Standard using a peer review process. In accordance with the guidance, peer reviews should also be used for PRA upgrades. A PRA upgrade is defined in the PRA Standard as "the incorporation into a PRA model of a new methodology or significant changes in scope or capability that impact the significant accident sequences or the significant accident progression sequences."

RG 1.174, Revision 3, "An Approach for Using Probabilistic Risk Assessment in Risk Informed Decisions on Plant-Specific Changes to the Licensing Basis," dated January 2018 (Reference 12), provides guidance on the use of PRA findings and risk insights in support of changes to a plant's licensing basis. This RG provides risk acceptance guidelines for evaluating the results of such evaluations.

3.0 TECHNICAL EVALUATION

3.1 NRC Staff's Method of Review

The NRC staff evaluated the licensee's application to determine if the proposed changes are consistent with the regulations and guidance discussed in Section 2.0 of this safety evaluation (SE). The NRC staff's review and the documentation of that review in this SE uses the framework of NEI 00-04, Revision 0.

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

Paragraph 50.69(b)(2)(i) of 10 CFR 50.69 states that a licensee voluntarily choosing to implement 10 CFR 50.69 shall submit an application for a 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." In addition, 10 CFR 50.69(c)(1)(v) states that the process for categorization must be "performed for entire systems and structures, not for selected components within a system or structure."

The guidance in RG 1.201, Revision 1, provides that the categorization process described in NEI 00-04, Revision 0, with any noted exceptions or clarifications, is acceptable for implementation of 10 CFR 50.69. Section 2 of NEI 00-04 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); and
8. SSC Categorization (Section 10 of NEI 00-04).

The guidance in RG 1.201, Revision 1, also states that the implementation of all processes described in NEI 00-04, Revision 0 (i.e., Section 2 through 12) is integral to providing reasonable confidence in the evaluations required by 10 CFR 50.69(c)(1)(iv) and that all aspects of NEI 00-04, must be followed to achieve reasonable confidence in the evaluations required by 10 CFR 50.69(c)(1)(iv).

The licensee stated in the LAR that it will implement the risk categorization process in accordance with NEI 00-04, Revision 0, as endorsed by RG 1.201 Revision 1. The LAR provided detail of the categorization process as follows: (1) summarization of the categorization process, (2) order of the sequence of elements or steps that will be performed (function/component level), (3) explanation of the difference between preliminary HSS and assigned HSS, and (4) identification of which inputs can and which cannot be changed by the IDP from preliminary HSS to LSS.

As summarized in the licensee's LAR, the categorization process contains the following elements/steps:

- Defining system boundaries (see Section 3.4 of this SE).
- Defining system function and assigning components to functions (see Section 3.4 of this SE).
- Risk Characterization. Safety-significance of active components is assessed through a combination of PRA and non-PRA methods, covering all hazards (see Section 3.5 of this SE).
- Defense-in-depth (DID) characterization performed in accordance with Section 6 of NEI 00-04, Revision 0 (see Section 3.6 of this SE).
- Passive Characterization. Passive components are not modeled in the PRA. Therefore, a different assessment method is used to assess the safety-significance of these components, as described in Section 3.5.4 of this SE. This process addresses those components that have only a pressure-retaining function and the passive function of active components, such as the pressure/liquid retention of the body of a motor operated valve.
    - Qualitative Characterization. System functions are qualitatively categorized as HSS or LSS based on the seven questions in Section 9.2 of NEI 00-04, Revision 0 (see Section 3.7 of this SE). The licensee confirmed in the response to PRA RAI 02 (Reference 3) that the IDP will independently determine that if any of the seven considerations cannot be confirmed for a function, then the final categorization of that function is HSS.
    - Cumulative risk sensitivity study. For PRA-modeled components, an overall risk sensitivity study is used to confirm that the population of LSS components results in acceptably small increases to CDF and LERF and meets the acceptance guidelines of RG 1.174 (see Section 3.8 of this SE).
    - Review by the IDP. The categorization results are presented to the IDP for review and approval. The IDP reviews the categorization results and makes the final determination on the safety-significance of system functions and components (see Section 3.9 of this SE).

In Table 3-1, "IDP Changes from Preliminary HSS to LSS," in Section 3.1.1, "Overall Categorization Process," of the LAR, the licensee provided details on how some steps of the process are performed at the component level (e.g., all PRA and non-PRA-modeled hazards, containment DID, passive categorization), how some steps are performed at the function level (e.g., qualitative criteria), and how some steps are performed at the function and component level (e.g., shutdown, core damage DID).

In Section 3.1.1 of the LAR, the licensee explained that consistent with NEI 00-04, Revision 0, the categorization of a component or function is "preliminary" until it has been confirmed by the IDP review (See also Section 3.9 of this SE). The LAR section includes a discussion and associated table (Table 3-1, repeated as Table 1, below) summarizing what mechanisms are available to assign functions and components to preliminarily HSS and how this designation can, or cannot, be changed to LSS by the IDP, as appropriate.

As illustrated in Table 1 below, all components that are assigned HSS based on risk significance as determined by the internal events importance measures, non-PRA risk models, DID considerations, passive categorization, or cumulative impact of the qualitative considerations must be assigned HSS. Components assigned preliminary HSS based on other PRA modeled risk results may be assigned LSS by the IDP. The licensee stated that a function is categorized as HSS if any if any component supporting the function is assigned HSS based on risk significance. The qualitative considerations generally categorize functions and if the IDP determines that any one of the seven considerations are false for a system function, then the function will be assigned HSS. Once a system function is identified as HSS, then all the components supporting that function are preliminary HSS and will be presented to the IDP for review. Many functions are design-basis functions and some components supporting those functions may not be necessary to achieve success of the safety-significant mitigating function, and these components may not be required to be HSS based on their risk significance. Therefore, any component supporting a HSS function, but not requiring a HSS assignment based on risk significance, may be assigned LSS by the IDP.

The NRC staff has evaluated the categorization steps and the associated clarifications provided by the licensee in the LAR and RAI responses and finds that the licensee's process is consistent with all aspects of the process in NEI 00-04 Revision 0, as endorsed by RG 1.201, Revision 1.

Table 1

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

Element	Categorization Step - NEI 00-04 Section	Evaluation Level	IDP Change HSS to LSS	Drives Associated Functions
Qualitative Criteria	Considerations – Section 9.2	Function	Allowable for Considerations	N/A
Passive	Passive – Section 4	Segment/ Component	Not Allowed	No

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

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. In addition, this step includes the critical evaluation of plant-specific risk information to ensure that the information is adequate to support this application. The guidance in Section 3 of NEI 00-04 summarizes the use of risk information and the general quality measures that should be applied to the risk analyses supporting the 10 CFR 50.69 categorization. These quality measures include characterization of technical acceptability of both the internal events at-power PRA and other risk analyses necessary to implement 10 CFR 50.69.

The licensee's risk categorization process uses PRAs to assess risks from internal events (including flooding), and from fire. For the other applicable hazard groups, the licensee's process uses non-PRA methods for the risk categorization. The licensee uses its Seismic Safe Shutdown Equipment List (SSEL) from the SMA to assess seismic risk, its IPEEE screening process to assess other external hazards (high winds and external floods), and its qualitative DID shutdown model to assess shutdown risk. The use of risk information and quality of the licensee's PRA is reviewed in Section 3.5 of this SE.

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

Paragraph 50.69(c)(1)(ii) of 10 CFR requires licensees to determine SSC functional importance using an integrated, systematic process for addressing initiating events (internal and external), SSCs, and plant operating modes, including those not modeled in the plant-specific PRA. The functions to be identified and considered include design-basis functions and functions credited for mitigation and prevention of severe accidents. Section 4 of NEI 00-04, Revision 0, 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, Revision 0, states that system selection and boundary definition include defining system boundaries where the system interfaces with other systems. The guidance in NEI 00-04 states that the next step is the identification of system functions, including design-basis and beyond-design-basis functions identified in the PRA, and that system functions should be consistent with the functions defined in design-basis documentation and maintenance rule functions. The guidance in NEI 00-04 states that the coarse mapping of components to functions involves the initial breakdown of system components into system functions they support. The licensee should then identify, and document system components and equipment associated with each function. The guidance in NEI 0-04 also includes consideration of interfacing functions. Section 7.1, "Engineering Categorization," of the NEI 00-04 guidance states in part, "[d]ue to the overlap of functions and components, a significant number of components support multiple functions. In this case, the SSC of Part thereof should be assigned the highest risk significance for any function that the SSC or Part thereof supports." Furthermore, Section 4 of the NEI 00-04 guidance states in part, "there may be circumstances where the categorization of a candidate low safety-significant SSC within the scope of the system being considered cannot be completed because it also supports an interfacing system. In this case, the SSC will remain uncategorized until the interfacing system is categorized."

The LAR summarizes the applicable guidance in NEI 00-04, Revision 0, and states that the guidance in NEI 00-04 will be followed. Therefore, the NRC staff finds that the licensee described a systematic process that will identify design-basis functions and functions credited for mitigation and prevention of severe accidents, consistent with the requirements of 10 CFR 50.69(c)(1)(ii), because all system functions will be identified and evaluated through the categorization process in accordance with NEI 00-04.

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

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

- Internal event risk
- Fire
- Seismic
- Other external risks (tornadoes, external floods)
- Shutdown risks

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

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

- SMA to assess seismic risk;
- IPEEE screening to assess the risk from other external hazards (e.g., high winds, external floods); and
- Shutdown Safety Plan to assess shutdown risk.

The methods used by the licensee to assess internal and external hazards are consistent with the methods included in the NEI 00-04, Revision 0 guidance, as endorsed by RG 1.201, Revision 1, and therefore, acceptable to the NRC staff. The guidance considers the results and insights from the plant-specific PRA peer reviews as required by 10 CFR 50.69(c)(1)(i), and non-PRA risk characterization as required by 10 CFR 50.69(c)(1)(ii). The application of these approaches is reviewed in the following SE subsections: PRA in Subsections 3.5.1 and 3.5.2, and the non-PRA methods in Subsection 3.5.3.

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

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

Paragraph 50.69(b)(2)(iii) of 10 CFR requires the results of the PRA review process conducted to meet 10 CFR 50.69(c)(1)(i) be submitted as part of the application. The licensee has submitted this information, and the NRC staff's review of this information is presented below.

In the LAR, the licensee did not state if Robinson has incorporated diverse and flexible coping strategies (FLEX) and associated equipment into its PRA models to support risk-informed decisionmaking in accordance with the guidance in RG 1.200, Revision 2. Therefore, the NRC staff requested in PRA RAI 08 (Reference 5) for the licensee to provide clarification on whether Robinson incorporates FLEX equipment and strategies into its PRA models. In response to PRA RAI 08 (Reference 3), the licensee confirmed that Robinson does not have FLEX equipment and strategies in its PRA models and therefore, no further review of FLEX modelling strategies was necessary.

3.5.1.1 Internal Events PRA

The NRC staff review of the internal events PRA was based on the results of the peer review of the internal events PRA performed in accordance with RG 1.200, Revision 2; the associated facts and observations (F&O) closure review described in Section 3.3, "PRA Review Process Results (10 CFR 50.69(b)(2)(iv))," of the LAR; and supplemented by previous docketed information on PRA quality submitted to the NRC for the integrated leak rate test interval license amendment dated November 19, 2015 (Reference 13) and the issuance of amendment to adopt NFPA 805 dated February 3, 2017 (Reference 14).

In the LAR, the licensee stated that the internal events model was subject to a self-assessment and a full scope peer review in May 2010 against RG 1.200, Revision 2, and the ASME/ANS 2009 Standard. In the LAR, the licensee further stated that in August 2017, an F&O closure review was performed by an independent team on all internal events finding-level F&Os. This F&O closure review was performed as detailed in Appendix X (Reference 15) to the guidance in NEI 05-04, "Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard" (Reference 16); NEI 07-12, Revision 1, "Fire PRA Peer Review Process Guidelines," dated June 2010 (Reference 17); and NEI 12-13 External Hazards PRA Peer Review Process Guidelines," dated August 2012 (Reference 18), concerning the process "Close-Out of Facts and Observations." The NRC staff accepted, with conditions, a final version of Appendix X to NEI 05-04, 07-12, and 12-13 in the NRC letter dated May 3, 2017

(Reference 19), and therefore, the final Appendix X guidance was available for the F&O closure review.

The licensee submitted a list of all the F&Os that remained open after the F&O closure review in Attachment 3, "Disposition and Resolution of Open Peer Review Findings and Self-Assessment Open Items," of the LAR. The licensee provided, for each F&O, a disposition for the F&Os for this application. The NRC staff reviewed the licensee's resolution of all the peer review findings and assessed the potential impact of the findings on the categorization. The NRC staff requested additional information to clarify the licensee's disposition for some of the findings as described in the following paragraph.

F&O LE-D5 indicated that human actions to isolate the ruptured steam generator following a steam generator tube rupture (SGTR) initiating event were not modeled. As reported in the LAR, the disposition stated that a sensitivity study demonstrates that the unmodeled human failure events (HFE) for isolating a ruptured steam generator following a SGTR initiating event has a minimal impact on the PRA results. Related F&Os AS-A5-03 and LE-D5-01 states, in part, that the thermally-induced SGTR accident sequence was missing from the Robinson PRA. The NRC staff requested clarification in PRA RAI 01.a (Reference 5) on whether the sensitivity study for the unmodeled HFE(s) for a SGTR initiating event included the thermally-induced SGTRs. In response to PRA RAI 01.a (Reference 3), the licensee stated that the human actions to isolate a SGTR event are not applicable to a thermally-induced SGTR. Given that the secondary side is faulted, the SG tube ruptures occur after core damage if the SG is faulted on the secondary side. This leaves little or no opportunity for operator actions to isolate the SG and prevent a large early release. The Fussell-Vesely or Risk Achievement Worth thresholds are risk metrics used to assess SSC importance. In the RAI response, the licensee also clarified that the sensitivity study revealed that including the previously unmodeled HFEs for the SGTR initiating events resulted in several basic events in the PRA now exceeding the Fussell-Vesely or Risk Achievement Worth thresholds for HSS in 10 CFR 50.69. As a result, the licensee will incorporate this operator action into the Robinson PRA model. If this update is determined to be a PRA model upgrade per the ASME/ANS 2009 Standard, the licensee will conduct a focused scope peer review prior to implement 10 CFR 50.69. The licensee provided Implementation Item ii (discussed in Section 3.5.5 of this SE) to incorporate this HFE into the internal events PRA model used for risk categorization. The NRC staff finds, that with the completion of Implementation Item ii, the licensee has adequately modeled HFEs for isolating a SGTR into its PRA to support the risk categorization process.

Paragraph 50.69(c)(1)(i) of 10 CFR, requires, in part, that any plant-specific PRA used in the categorization must be of sufficient quality and level of detail to support the categorization process, and must be subjected to a peer review process assessed against a standard that is endorsed by the NRC. RG 1.200, Revision 2, provides guidance for determining the technical adequacy of an internal events PRA by comparing the PRA to the relevant parts of the ASME/ANS 2009 Standard using a peer review process. Based on its review, the NRC staff finds that the licensee has followed the guidance in RG 1.200 and submitted the results of the peer review, and therefore, meets the requirement in 10 CFR 50.69(b)(2)(iii).

The NRC staff has reviewed the peer review results and finds that the quality and level of detail of the internal events PRA is sufficient to support the categorization of SSCs, as required by 10 CFR 50.69(b)(2)(ii), using the process endorsed by the NRC staff in RG 1.201, Revision 1. Significant errors and weaknesses with the internal events PRA will be resolved with the completion of Implementation Item ii (discussed in Section 3.5.5 of the SE). Therefore, the NRC staff concludes that the quality of the internal events PRA with the completion of the Implementation Item ii, meets the requirement in 10 CFR 50.69(c)(1)(i).

3.5.1.2 Internal Flooding PRA

The internal flooding PRA was subject to a self-assessment and a full scope peer review in August 2015 against RG 1.200 Revision 2 and the ASME/ANS 2009 Standard. In August 2017, an F&O closure review was performed by an independent team on all internal flooding finding-level F&Os.

The licensee submitted a list of the F&Os that remained open after the F&O closure review as described in LAR Attachment 3. The licensee provided a disposition for each F&O for this application. The NRC staff reviewed the licensee's resolution of all the peer review findings and assessed the potential impact of the findings on the categorization. The NRC staff requested additional information to clarify the licensee's disposition for some of the findings as described in the following paragraphs.

F&O IFEV-A7-01, regarding human-induced flooding events screening, states that the screening out of all human-induced flooding events may have been non-conservative. The disposition for F&O IFEV-A7-01 states that a sensitivity study was performed but did not provide a description of the sensitivity study or the results and insights from the sensitivity study. In PRA RAI 01.b (Reference 5), the NRC staff requested the details of the sensitivity and how the results and insights of the sensitivity study impact the risk categorization process, as well as justification for why the human-induced flooding events were screened. In response to PRA RAI 01.b (Reference 3), the licensee stated that the sensitivity study provided no risk insights in terms of the impact of human-induced flooding. As a result, the licensee provided Implementation Item i (discussed in Section 3.5.5 of this SE) to evaluate and incorporate human-induced flood events into the internal flooding PRA model used for risk categorization by using an industry approved method prior to implementing 10 CFR 50.69. If this update is determined to be a PRA model upgrade per the ASME/ANS 2009 Standard, the licensee will conduct a focused scope peer review prior to implementation of 10 CFR 50.69. The NRC staff finds, with the completion of Implementation Item i, that the licensee has adequately screened human-induced flooding events and has incorporated the events that did not screen into its internal flooding PRA. The staff finds that it is therefore adequate to support the risk categorization process.

F&O IFSN-A8-01 states that the use of the Electric Power Research Institute (EPRI) door failure criteria may not be appropriate depending on the characteristics of the door and the flooding scenario. The licensee stated in its disposition that the majority of components would fail at the door failure criteria, and these criteria would have minimal effects on modeling and have no impact to the risk categorization process. In PRA RAI 01.c (Reference 5), the NRC staff requested the justification for the exclusion of the correct door failure heights and that there would be no impact on the risk categorization process. In response to PRA RAI 01.c (Reference 3), the licensee stated that the doors at Robinson are typical of nuclear power plant doors, and any difference between the generic EPRI door failure heights and the site-specific door failure heights is small. The small difference in door height would result in negligible differences in the calculated isolation failure probability. The NRC staff finds that the EPRI door failure criteria should be representative of the Robinson doors due to the licensee stating that it has typical doors. Therefore, the door failure modes are modeled adequately with respect to the expected contribution of flooding induced door failure to risk and are consistent with the resolution described in flooding PRA methods documents.

F&O IFSN-A8-02 identifies one scenario where additional equipment may be impacted by a flood propagating from a door gap. The licensee's disposition states that crediting flow underneath door gaps would increase the operator's time to isolate a flooding scenario, and therefore, the modeling is conservative. In PRA RAI 01.d (Reference 5), the NRC staff requested justification that the exclusion of the additional equipment impacts does not affect any SSC risk categorization. In response to PRA-RAI 01.d (Reference 3), the licensee stated that the only area with additional equipment that may be impacted by a flood propagating from a door gap is a large open area that would not allow for water to accumulate to a depth that would impact the SSCs. The NRC staff finds that the equipment in this area can be excluded due to water not being able to accumulate from the door gap flow, and there is no expected impact on the risk categorization process.

RG 1.200, Revision 2, provides guidance for determining the technical adequacy of an internal flooding PRA by comparing the PRA to the relevant parts of the ASME/ANS 2009 Standard using a peer review process. Based on its review, the NRC staff finds that the licensee has followed the guidance in RG 1.200 and submitted the results of the peer review, and therefore, meets the requirement in 10 CFR 50.69(b)(2)(iii). The NRC staff has reviewed the peer review results and finds that the quality and level of detail of the internal flooding PRA is sufficient to support the categorization of SSCs, as required by 10 CFR 50.69(b)(2)(ii), using the process endorsed by the NRC staff in RG 1.201, Revision 1. Significant errors and weaknesses with the internal flooding PRA will be resolved with the completion of Implementation Item i (discussed in Section 3.5.5 of the SE). Therefore, the NRC staff concludes that the quality of the internal flooding PRA with the completion of the Implementation Item i, meets the requirement in 10 CFR 50.69(c)(1)(i).

3.5.1.3 Fire PRA

The licensee's fire PRA was subject to a full-scope industry peer review in May 2013 against RG 1.200, Revision 2. In August 2017, an F&O closure review was performed by an independent team on all fire finding-level F&Os.

The licensee submitted a list of the F&Os that remained open after the F&O closure review in LAR Attachment 3. The licensee provided, for each F&O, a disposition for the F&Os for this application. The NRC staff reviewed the licensee's resolution of all the peer review findings and assessed the potential impact of the findings on the categorization.

In Attachment 6, "Disposition and Resolution of Open Peer Review Findings and Self-Assessment Open Items," of the LAR, the licensee stated that the current method of crediting incipient detection is similar to that found in NUREG-2180, "Determining the Effectiveness, Limitations, and Operator Response for Very Early Warning Fire Detection Systems in Nuclear Facilities (DELORES-VEWFIRE)," dated December 2016 (Reference 20). The NRC staff notes that the letter to the industry dated July 1, 2016 (Reference 21), superseded the guidance of an earlier method (FAQ 08-0046) to NUREG-2180. Since the LAR did not provide details on how the licensee's method of crediting incipient detection is similar to that found in NUREG-2180, it is unclear to the NRC if there are any differences between the licensee's approach and NUREG-2180, or if the licensee is using methods from superseded guidance. In PRA RAI 04 (Reference 5), the NRC staff requested the licensee to justify that its methodology is acceptable for use in the risk categorization process. In response to PRA RAI 04, the licensee clarified that the methodology used for crediting incipient detection is described in NUREG-2180. Therefore, the NRC staff finds the licensee's methodology for crediting incipient detection to be acceptable because it uses the NRC accepted guidance in NUREG-2180.

Paragraph 50.69(c)(1)(i) of 10 CFR, requires, in part, that any plant-specific PRA used in the categorization must be of sufficient quality and level of detail to support the categorization process and must be subjected to a peer review process assessed against a standard that is endorsed by the NRC. RG 1.200, Revision 2, provides guidance for determining the technical adequacy of a fire PRA by comparing the PRA to the relevant parts of the ASME/ANS 2009 Standard using a peer review process. Based on its review, the NRC staff finds that the licensee has followed the guidance in RG 1.200 and submitted the results of the peer review, and therefore, meets the requirement in 10 CFR 50.69(b)(2)(iii). The NRC staff has reviewed the peer review results and finds that the quality and level of detail of the fire PRA is sufficient to support the categorization of SSCs, as required by 10 CFR 50.69(b)(2)(ii), using the process endorsed by the NRC staff in RG 1.201, Revision 1. Therefore, the NRC staff concludes that the quality of the fire PRA meets the requirement in 10 CFR 50.69(c)(1)(i).

3.5.2 Importance Measures and Sensitivity Studies

Paragraph 50.69(c)(1)(i) of 10 CFR requires the licensee to consider the results and insights from the PRA be used during categorization. These requirements are met, in part, by using importance measures and sensitivity studies, as described in the methodology in NEI 00-04, Section 5.

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

The guidance in NEI 00-04, Revision 0, specifies sensitivity studies to be conducted for each PRA model. The sensitivity studies are performed to ensure that assumptions associated with these specific uncertain parameters (i.e., human error, common cause failure, and maintenance probabilities) are not masking the importance of a component. The NEI 00-04 guidance states that any additional "applicable sensitivity studies" from characterization of PRA adequacy should be considered. LAR Section 3.2.7, "PRA Uncertainty Evaluations," describes how the licensee searched for additional issues in the internal events (including internal flooding) PRA that should be evaluated with a sensitivity study. The licensee used the NRC guidance in NUREG-1855, Revision 0, "Guidelines on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making," dated March 2009 (Reference 22), supplemented with the EPRI Technical Report (TR)-1016737, "Treatment of Parameter and Modeling Uncertainty for Probabilistic Risk Assessments," dated December 2008 (Reference 23), to identify sources of uncertainty in the internal events PRA.

The NRC staff noted in PRA RAI 03.a (Reference 5) that NUREG-1855, Revision 0, which was referenced in the LAR, has been superseded by NUREG-1855, Revision 1 (Reference 24), which References the additional EPRI guidance (EPRI TR-1026511, "Practical guidance on the Use of Probabilistic Risk Assessment in Risk Informed Applications with a Focus on the Treatment of Uncertainty," dated December 2012 (Reference 25)). The original EPRI guidance

(TR-1016737) provided a generic list of assumptions for internal events PRA, while the additional EPRI guidance (TR-1026511) added generic assumptions for external events (e.g., fire PRAs). The guidance in NUREG 1855 suggests that any assumption/uncertainties identified during the peer reviews should also be addressed. In PRA RAI 03.a (Reference 5) and RAI 3.01 (Reference 6), the licensee described the process used to determine the candidates for key sources of uncertainty, and how this process either demonstrates consistency with NUREG-1855, Revision 1, or is technically acceptable and adequate to identify and disposition the key sources of uncertainty to support the categorization process.

In the response to PRA RAI 03.b (Reference 3) and RAI 3.01 (Reference 4), the licensee stated that, consistent with NUREG-1855, Revision 1, Stage E, it used Table A.1 of EPRI 1016737 as well as the PRA documentation for plant-specific assumptions and uncertainties to identify the assumptions and uncertainties used in the internal events and internal flood base PRA models supporting the categorization. For assumptions and uncertainties used in the fire base PRA model supporting the categorization, the licensee reviewed the generic issues identified in EPRI TR-1026511, as well as the PRA documentation for plant-specific assumptions and uncertainties. The responses to PRA RAI 03.b and RAI 3.01 summarized the considerations used to determine whether each assumption and uncertainty was key to this application or not. The licensee stated that the considerations were based on the definitions in RG 1.200, Revision 2; NUREG-1855, Revision 1; and related References (i.e., EPRI TR-1016737, EPRI TR-1013491, "Guideline for the Treatment of Uncertainty in Risk-Informed Applications," dated October 2006 (Reference 26) and EPRI TR-1026511). The NRC staff finds that the identified considerations are consistent with the referenced documents and therefore, provide a reasonable basis for the evaluation.

In Attachment 6 of the LAR, the licensee provided the list of assumptions and sources of modeling uncertainty that may be key for this application. The NRC staff found that the dispositions for two of the assumptions and modeling uncertainties reported in the LAR could impact the categorization process. Therefore, the NRC staff requested, in PRA RAI 05 and RAI 5.01, for the licensee to address the dispositions of the assumptions for feed and bleed success criteria and joint human error probabilities.

In response to PRA RAI 05.a (Reference 3) and RAI 5.01 (Reference 4) regarding the feed and bleed success criteria given the loss of secondary heat removal, the licensee stated that late loss of secondary heat removal (i.e., 50 minutes) would significantly reduce the success criteria requirements. In a sensitivity study that reduced the success criteria for feed and bleed from two power operated relief valves to one power operated relief valve, the results showed no basic events that were originally LSS became HSS. From the results of the licensee's sensitivity study, the NRC staff concludes that the 10 CFR 50.69 categorization is not sensitive to the success criteria for feed and bleed. Therefore, the NRC staff finds the exclusion of the refined success criteria for feed and bleed to be acceptable because it does not affect any of the SSC risk categorizations.

In response to PRA RAI 05.b, regarding whether joint human error probabilities floor values were used in the internal events PRA, the licensee clarified there are no dependency combinations in the internal events PRA below the suggested floor value of 1E-6. The NRC staff finds this response to be acceptable for this application because it is consistent with the guidance in NUREG-1792, "Good Practices for Implementing Human Reliability Analysis (HRA)," dated April 2005 (Reference 27).

In Attachment 6 of the LAR, the licensee identified three issues: the reactor coolant pump seal failure, loss of offsite power frequencies, and fire modeling uncertainties and dispositioned the issues by referring to the sensitivity studies in Tables 5-2, 5-3, 5-4, and 5-5 in NEI 00-04, Revision 0. In PRA RAI 06 (Reference 5), the NRC staff noted that the NEI 00-04 tables did not refer to any of the three issues and requested clarification. In the response to PRA RAI 06 (Reference 3), the licensee referred to its updated uncertainty analysis described in the response to PRA RAI 03 and RAI 3.01 and provided an updated table in the letter dated May 6, 2019 (Reference 4), of the disposition of key assumptions/sources of uncertainty, which superseded Attachment 6 in the LAR. The updated table in the May 6, 2019, letter, does not include two issues and identifies one specific fire modeling issue instead of general fire modeling methods. The NRC staff determined that the acceptability of the disposition of these issues relies on the acceptability of the licensee's process to evaluate uncertainties described in its response to PRA RAI 03 and followup RAI 3.01, which is discussed in the preceding paragraphs.

The different key assumptions and uncertainties identified in Attachment 6 of the LAR and in the licensee responses to PRA-RAI-03 (Reference 5) and RAI 3.01 (Reference 4) correspond to the changing approach the licensee used to identify potential key assumptions and sources of uncertainties. In addition to other changes, such as removing or redefining the key assumptions and uncertainties associated with reactor coolant pump seal failure, loss-of-offsite power frequencies, and fire modeling discussed in PRA RAI 06, as well as the updated uncertainty analysis in response to RAI 3.01, the licensee identified a previously unidentified issue related to failure to model human actions to isolate the accumulators to allow switchover to shutdown cooling in some scenarios. The licensee determined that this missing modeling could affect the categorization of some SSCs and provided Implementation Item iii to develop and incorporate these human actions into the PRA models used for risk categorization.

Given the licensee's assessment and its response, the NRC staff finds that the licensee searched for, identified, and evaluated sources of uncertainty in its PRA models consistent with the guidance in RG 1.200, Revision 2, NUREG-1855, Revision 1, and EPRI document TR-1016737. Therefore, the NEI 00-04 guidance to identify additional "applicable sensitivity studies" is satisfied.

3.5.3 Non-PRA Methods

According to 10 CFR 50.69(c)(1)(ii), 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.

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

- SMA to assess seismic risk;
- Screening during the IPEEE to assess risk from other external hazards;
- Shutdown Safety Plan as described in NUMARC 91-06 (Reference 8) to assess shutdown risk.

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

Seismic Risk

To assess seismic risk for the 10 CFR 50.69 categorization process, the licensee proposes to use the SMA method. SMA is a screening method that does not quantify CDF. The licensee used the SMA method during its IPEEE in response to GL 88-20 (References 28 and 29). The licensee stated in Section 3.2.3, "Seismic Hazards," of the LAR (Reference 1) that it will follow the NEI 00-04 (Reference 7) approach using the SSEL to identify credited equipment as HSS regardless of its capacity, frequency of challenge or level of functional diversity. The licensee stated in the LAR that it had conducted an updated evaluation of the SMA SSEL to reflect the current as-built and as-operated plant. In addition, the licensee stated that future changes to the plant will be evaluated as needed to determine their impact on the SMA and risk categorization process.

Consistent with NEI 00-04, Revision 0, the licensee's 10 CFR 50.69 categorization process considers all components in the SSEL as HSS based on seismic risk. All components not listed in the SSEL are considered preliminary LSS with respect to seismic risk.

The method proposed by the licensee to assess seismic risk is consistent with the NRC-endorsed methods in NEI 00-04, Revision 0, and therefore, the NRC staff finds it acceptable for use in the licensee's 10 CFR 50.69 categorization process.

Other External Hazards

As indicated in the LAR, external hazards were initially evaluated by the licensee during the IPEEE. This hazard category includes all non-seismic external hazards such as high winds, external floods, transportation and nearby facility accidents, and other hazards. The IPEEE external hazard analysis used a progressive screening approach and concluded that all these other hazards are negligible contributors to overall plant risk. Further, the licensee indicated that it had reevaluated these other external hazards using the criteria in the ASME/ANS 2009 Standard (Reference 11).

In Section 3.2.4, "Other External Hazards," of the LAR, the licensee states that an evaluation is performed to determine if there are SSCs 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, these components would be considered HSS. Section 3.2.4 concludes that all remaining hazards were screened from applicability and considered insignificant for every SSC and, therefore, will not be considered during the categorization process.

The conclusion of LAR Section 3.2.4 implies that the external hazard assessment has been completed (based on the IPEEE) and that all external hazards will never need evaluation during categorization. In PRA RAI 07 (Reference 5), the NRC staff requested that the licensee clarify when and how it is determined that the failure of SSCs would result in an unscreened scenario. In response to PRA RAI 07 (Reference 3), the licensee clarified that, "Per NEI 00-04, the external hazard assessment is required for each SSC categorization. As such, each SSC being categorized will be assessed in accordance with NEI 00-04 Figure 5-6 for the external hazards... If the failure of the SSC results in the screening criterion from Attachment 5 [of the LAR] not being met, then the scenario would become unscreened and the SSC would become candidate High Safety Significant." The NRC staff finds this clarification acceptable for the application because it is consistent with the guidance in Section 5.4, "Assessment of Other External Hazards," of NEI 00-04, Revision 0.

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

Shutdown Risk

Consistent with the guidance in NEI 00-04, Revision 0, the licensee proposes to use the shutdown safety assessment process based on NUMARC 91-06 (Reference 8). NUMARC 91-06 provides considerations for maintaining DID for the five key safety functions during shutdown, namely, decay heat removal capability, inventory control, power availability, reactivity control, and containment - primary/secondary. The guidance in NUMARC 91-06 specifies that a DID approach should be used with respect to each defined shutdown key safety function. This is accomplished by designating a running and an alternative system/train to accomplish the given key safety function.

The licensee's process is consistent with the guidance in NEI 00 04, Revision 0, Section 5.5, "Shutdown Safety Assessment." The licensee indicated that components are categorized with respect to shutdown risk using a non-PRA shutdown assessment as follows:

- If a system/train supports a key safety function as the primary or first alternate means, then it is considered to be a "primary shutdown safety system" and is categorized as preliminary HSS. NEI 00-04 defines a "primary shutdown safety system" as also having the following attributes:

    - It has a technical basis for its ability to perform the function.
    - It has margin to fulfill the safety function.
    - It does not require extensive manual manipulation to fulfill its safety function.

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

As explained above, the shutdown safety assessment method proposed by the licensee is consistent with the guidance in NEI 00-04, Revision 0. In addition, the method meets 10 CFR 50.69(c)(1)(ii) by using an integrated and systematic process that could identify HSS components if they existed, consistent with the shutdown evaluation process, as described in the NRC-endorsed NEI 00-04. Therefore, the NRC staff finds the licensee's proposed method acceptable.

3.5.4 Component Safety Significance Assessment for Passive Components

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

In the LAR, the licensee proposed using a categorization method for passive components not cited in NEI 00-04 for passive component categorization, but approved by the NRC on April 22, 2009, for Arkansas Nuclear One, Unit 2 (ANO-2) (Reference 30). The ANO-2 methodology is a risk-informed safety classification and treatment program for repair/replacement activities for

Class 2 and Class 3 pressure-retaining items and their associated supports (exclusive of Class concrete containment and metallic containment items), using a modification of the ASME Code Case N-660, "Risk-Informed Safety Classification for Use in Risk-Informed Repair/Replacement Activities, Section XI, Division 1," dated July 2002 (Reference 31). The ANO 2 methodology relies on the conditional core damage and large early release probabilities associated with pipe ruptures. Safety-significance is generally measured by the frequency and the consequence of, in this case, pipe or other passive pressure boundary ruptures. Treatment requirements (including repair/replacement) only affect the frequency of passive component failure. Categorizing solely based on consequences, which measures the safety-significance of the pipe given that it ruptures, is conservative compared to including the rupture frequency in the categorization. The categorization will not be affected by changes in frequency arising from changes to the treatment.

In Section 3.1.2 "Passive Categorization Process," of the LAR, the licensee stated that all ASME Class 1 SSCs with a pressure retaining function, as well as supports, will be assigned HSS. The licensee will apply the ANO-2 methodology to ASME Class 2 and 3 SSCs only. Therefore, the NRC staff finds that the use of the repair/replacement methodology is acceptable because it will only be used for passive component categorization of Class 2 and Class 3 SSCs.

3.5.5 Summary

The NRC staff reviewed the PRA and the non-PRA methods used by the licensee in its 10 CFR 50.69 categorization process to assess the safety-significance of active and passive components and finds these methods acceptable and consistent with RG 1.201, Revision 1, and the NRC-endorsed guidance in NEI 00-04, Revision 0. Accordingly, subject to the proposed license condition described below, the NRC staff approves the use of the following methods in the licensee's 10 CFR 50.69 categorization process:

- PRA to assess internal events, including internal flooding risk
- Fire PRA to assess fire risk
- SMA to assess seismic risk
- Screening using IPEEE to assess risk from other external hazards (high winds, external floods)
- Shutdown safety assessment process to assess shutdown risk
- ANO-2 (Reference 30) passive categorization method to assess passive component risk for Class 2 and Class 3 SSCs and their associated supports

Based on its review of the LAR and the licensee's responses to the staff's RAIs, the NRC staff identified certain specific actions necessary to support its conclusion that the proposed program meets the requirements in 10 CFR 50.69 and the guidance in RG 1.201 and NEI-00-04. The licensee proposed the addition of a license condition for the implementation of 10 CFR 50.69 (see Section 4.0 of this SE). Specifically, the license condition states that Duke Energy will complete the prerequisite items. The NRC staff's evaluation of the proposed license condition is in Section 4.0 of this SE.

The licensee provided the following implementation items in Attachment 1 of its RAI response dated May 6, 2019 (Reference 4):

i. Update the Robinson internal flood model to account for generic human induced flooding events using an industry accepted methodology.

ii. Update the Robinson internal events model to include HFE related to isolating a ruptured steam generator following a SGTR. If this update is determined to be a PRA model upgrade per the ASME/ANS 2009 PRA Standard, then Robinson will conduct a focused scope peer review.

iii. Update the Robinson internal events, internal flood, and fire PRA models to account for isolation of the reactor coolant system accumulators.

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

Paragraph 50.69(c)(1)(iii) of 10 CFR requires that the process used for categorizing SSCs must maintain DID. NEI 00-04, Revision 0, Section 6, provides guidance on assessment of DID. In Section 3.1.1 of the LAR, the licensee stated that it will require an SSC categorized as HSS based on the DID assessment in Section 6 to be categorized as HSS.

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

RG 1.201, Revision 1, endorses the guidance in NEI 00-04, Section 6 but notes that the containment isolation criteria in this section of NEI 00-04 are separate and distinct from those set forth in 10 CFR 50.69(b)(1)(x). The criteria in 10 CFR 50.69(b)(1)(x) are to be used in determining which RISC 3 (i.e., safety-related LSS) containment penetrations and valves may be exempted from the Type B and Type C leakage testing requirements in both Options A and B of Appendix J to 10 CFR Part 50, but the 10 CFR 50.69(b)(1)(x) criteria are not used to determine the RISC category for containment isolation valves or penetrations.

Based on the NRC staff's review of the licensee's categorization process, the staff finds that the licensee's process is consistent with the NRC-endorsed NEI 00-04 guidance and fulfills the 10 CFR 50.69(c)(1)(iii) criteria that DID is maintained.

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

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

In LAR Section 3.1.1, the licensee stated that if any component is identified as HSS from either the integrated risk component safety-significance assessment (Section 5 of NEI 00-04), the DID assessment (Section 6 of NEI 00-04), or the Qualitative Criteria (Section 9 of NEI 00-04), the associated system function(s) would be identified as HSS. Once a system function is identified as HSS, then all the components that support that function are categorized as preliminary HSS.

The NRC staff finds that the default assignment of LSS to functions associated with components that have been assigned HSS by non-PRA deterministic methods is consistent with NEI 00-04, Revision 0, and therefore, acceptable.

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

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

This sensitivity study, together with the periodic review process discussed in Section 3.10 of this SE, assure that the potential cumulative risk increase from the categorization is small. The performance monitoring process monitors the component performance to ensure that potential increases in failure rates of categorized components are detected and addressed before reaching the rate assumed in the sensitivity study.

The NRC staff finds that the licensee will perform the risk sensitivity study consistent with the guidance in NEI 00-04, Section 8.0, and therefore, will assure that the potential cumulative risk increase from the categorization is small, as required by 10 CFR 50.69(c)(1)(iv).

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

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

The guidance in NEI 00-04, Revision 0, endorsed in RG 1.201, Revision 1, ensures that the IDP expertise is sufficient to perform the categorization and that the results of the different evaluations (PRA and non-PRA) are used in an integrated, systematic process, as required by 10 CFR 50.69(c)(1)(ii). As provided by the NEI 00-04 guidance, and as indicated in LAR Attachment 1, the process used by the IDP for the categorization of SSCs will be described and documented in a plant procedure.

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

Based on its review, the NRC staff finds that the licensee's IDP areas of expertise meet the requirements in 10 CFR 50.69(c)(2), and the additional descriptions of the IDP characteristics, training, processes, and decision guidelines are consistent with NEI 00-04, Revision 0, as endorsed by RG 1.201, Revision 1. Therefore, all aspects of the integrated, systematic process used to characterize SSCs will reasonably reflect current plant configuration and operating practices, and applicable plant and industry operational experience as required by 10 CFR 50.69(c)(1)(ii).

Paragraph 50.69(c)(1)(iv) of 10 CFR requires, in part, reasonable confidence that sufficient safety margins are maintained for SSCs categorized as RISC-3. The licensee addresses safety margins through an integrated engineering evaluation that would nominally be addressed by the IDP. As discussed in the guidance in NEI 00-04, Revision 0, endorsed by RG 1.201, Revision 1, the IDP need not explicitly consider safety margins. Sufficient safety margin will be maintained because the RISC-3 SSCs will remain capable of performing their safety-related functions as required by 10 CFR 50.69(d)(2), and because any potential increases in CDF and LERF that might stem from changes in RISC–3 SSC reliability due to reduced treatment permitted by 10 CFR 50.69 will be maintained small, as required by 10 CFR 50.69(c)(1)(iv). Therefore, the NRC staff finds that the program implemented by the licensee, consistent with the endorsed guidance in NEI 00-04, fulfills the 10 CFR 50.69(c)(1)(iv) criteria that sufficient safety margins are maintained.

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

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

Section 50.69(e) of 10 CFR requires periodic updates to the licensee's PRA and SSC categorization. As provided in RG 1.200, Revision 2, the NRC staff review of the PRA quality and level of detail reported in this SE is based primarily on determining how the licensee has resolved key assumptions and areas identified by peer reviewers as being of concern (i.e., F&Os). As discussed above in this SE, the implementation items that are included as a license condition will be completed prior to implementation of the 10 CFR 50.69 categorization process, and will properly address items which, absent the license condition, could have a substantive impact on the PRA results.

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

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

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

In addition, LAR Attachment 1, "List of Categorization Prerequisites," states that the licensee will establish procedures for the use of the categorization process that contains the following elements: (1) IDP member qualification requirements, (2) qualitative assessment of system functions, (3) component safety-significance assessment, (4) assessment of DID and safety margin, (5) review by the IDP and final determination of safety-significance for system functions and components, (6) risk sensitivity studies to confirm that the risk acceptance guidelines of RG 1.174 are met, (7) periodic reviews to ensure continued categorization validity and acceptable performance for SSCs that have been categorized, and (8) documentation requirements identified in LAR Section 3.1.1. Procedures are formal plant documents, and changes will be tracked providing change control and records of the changes.

These categorization documents and records, as described by the licensee, include documentation and record change controls consistent with NEI 00-04, Revision 0, and endorsed by RG 1.201, Revision 1, and are in conformance with the requirements of 10 CFR 50.69(f)(1). Therefore, the NRC staff finds the documentation and records acceptable.

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

3.11 Technical Conclusion

The NRC staff reviewed the licensee's 10 CFR 50.69 categorization process and concludes that the licensee adequately implements 10 CFR 50.69 using models, methods, and approaches consistent with NEI 00-04, Revision 0, and RG 1.201, 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 acceptable for categorizing the safety significance of SSCs. Specifically, the staff concludes that the licensee's categorization process:

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

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

(3) Maintains DID, as reviewed in Section 3.6 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 Sections 3.8 and 3.9 of this SE, and therefore, meets the requirements in 10 CFR 50.69(c)(1)(iv).

(5) Is performed for entire systems and structures, rather than for selected components within a system or structure, as reviewed in Section 3.3 of this SE, and therefore, the requirements in 10 CFR 50.69(c)(1)(v) will be met upon implementation.

(6) Includes categorization by IDP, staffed with expert, plant-knowledgeable members whose expertise includes, at a minimum, PRA, safety analysis, plant operation, design engineering and system engineering, as reviewed in Section 3.9 of this SE, and therefore, meets the requirements in 10 CFR 50.69(c)(2).

4.0 10 CFR 50.69 IMPLEMENTATION LICENSE CONDITION

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 license application if it determines that the categorization process satisfies the requirements of 10 CFR 50.69(c). As described in this SE, the NRC staff has concluded that the 10 CFR 50.69 categorization process described in the licensee's application, as supplemented, includes a description of the categorization process that satisfies the requirements of 10 CFR 50.69(c). However, based on its review of the LAR and the licensee's responses to the NRC staff's RAIs, the NRC staff identified certain specific actions, as described below, that are necessary to support the staff's conclusion that the proposed program meets the requirements in 10 CFR 50.69 and the guidance in RG 1.201, Revision 1 and NEI 00-04, Revision 0.

The NRC staff's finding on the acceptability of the PRA evaluation in the licensee's proposed 10 CFR 50.69 process is conditioned on the completion of three changes to the PRA model. The three PRA model changes are identified as "Robinson 50.69 PRA Implementation Items" in Attachment 1 of the licensee's letter dated May 6, 2019 (Reference 4). The licensee and the NRC staff note that the implementation items are required to be completed prior to the implementation of 10 CFR 50.69 at Robinson.

In Attachment 2 of the letter dated May 6, 2019 (Reference 4), the licensee proposed the following condition to its license:

> Duke is approved to implement 10 CFR 50.69 using the processes for categorization of Risk Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 structures, systems, and components (SSCs) using: Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events, including internal flooding, internal fire, high winds, and external flood; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 SSCs and their associated supports; and the results of non PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards, i.e., seismic margin analysis (SMA) to evaluate seismic risk, and a screening of other external hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA Sa 2009; as specified in Unit 2 License Amendment No. 266 dated September 24, 2019.
>
> Duke Energy will complete the implementation items listed in Attachment 1 of Duke letter to NRC dated May 6, 2019 prior to implementation of 10 CFR 50.69. All issues identified in the attachment will be addressed and any associated changes will be made, focused-scope peer reviews will be performed on changes that are PRA upgrades as defined in the PRA standard (ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, Revision 2), and any findings will be resolved and reflected in the PRA of record prior to implementation of the 10 CFR 50.69 categorization process.
>
> Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from a seismic margins approach to a seismic probabilistic risk assessment approach).

Duke provided the following implementation items in Attachment 1 of its RAI response dated May 6, 2019 (Reference 4):

i. Update the Robinson internal flood model to account for generic human induced flooding events using an industry accepted methodology.

ii. Update the Robinson internal events model to include HFEs related to isolating a ruptured steam generator following a SGTR. If this update is determined to be a PRA model upgrade per the ASME/ANS 2009 PRA standard, then Robinson will conduct a focused scope peer review.

iii. Update the Robinson internal events, internal flood, and fire PRA models to account for isolation of the RCS accumulators.

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

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the NRC staff notified the South Carolina State official on August 5, 2019, of the proposed issuance of the amendment. The State official expressed no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes requirements 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 amendment involves 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 amendment involves no significant hazards consideration, published in the *Federal Register* on June 5, 2018 (83 FR 26101), and there has been no public comment on such finding. Accordingly, the amendment meets 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 amendment.

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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

8.0 REFERENCES

1. Kapopoulos, E. J., Jr., Duke Energy, letter to U. S. Nuclear Regulatory Commission, H. B. Robinson Steam Electric Plant, Unit No. 2, Renewed Facility Operating License Nos. DPR-23, NRC Docket No. 50-261, "Application to Adopt 10 CFR 50.69, 'Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors,'" dated April 5, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18099A130).

2. Kapopoulos, E. J., Jr., Duke Energy, letter to U. S. Nuclear Regulatory Commission, H. B. Robinson Steam Electric Plant, Unit No. 2, Renewed Facility Operating License Nos. DPR-23, NRC Docket No. 50-261, "Supplement to the Application to Adopt 10 CFR 50.69, 'Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors,'" dated June 6, 2018 (ADAMS Accession No. ML18162A147).

3. Donahue, J., Duke Energy, letter to U. S. Nuclear Regulatory Commission, H. B. Robinson Steam Electric Plant, Unit No. 2, Renewed Facility Operating License Nos. DPR-23, NRC Docket No. 50-261, "Response to NRC Request for Additional Information (RAI) Regarding Application to Adopt 10 CFR 50.69, 'Risk-Informed Categorization and Treatment of Structures, Systems, and Components (SSCs) for Nuclear Power Reactors,'" dated November 13, 2018 (ADAMS Accession No. ML18317A026).

4. Kapopoulos, E. J., Jr., Duke Energy, letter to U. S. Nuclear Regulatory Commission, H. B. Robinson Steam Electric Plant, Unit No. 2, Renewed Facility Operating License Nos. DPR-23, NRC Docket No. 50-261, "Response to NRC Request for Additional Information (RAI) Regarding Application to Adopt 10 CFR 50.69, 'Risk-Informed Categorization and Treatment of Structures, Systems, and Components (SSCs) for Nuclear Power Reactors,'" dated May 6, 2019 (ADAMS Accession No. ML19126A143).

5. Galvin, D., U.S. Nuclear Regulatory Commission, email to A. Zaremba, Duke Energy, "Robinson Draft RAIs – LAR to Allow Implementation of the Provisions 10 CFR 50.69 (EPID L-2018-LLA-0095) and LAR to Adopt TSTF-425 (EPID L-2018-LLA-0104)," dated September 28, 2018 (ADAMS Accession No. ML18274A263).

6. Jordan, N., U.S. Nuclear Regulatory Commission, email to A. Zaremba, Duke Energy, "H. B. Robinson Nuclear Power Plant, Unit 2 – 50.69 LAR RAIs (L-2018-LLA-0095)," dated March 12, 2019 (ADAMS Accession No. ML19072A024).

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

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

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

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

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Date: September 24, 2019

SUBJECT: H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2 – ISSUANCE OF AMENDMENT NO. 266 TO ADOPT 10 CFR 50.69, "RISK-INFORMED CATEGORIZATION AND TREATMENT OF STRUCTURES, SYSTEMS, AND COMPONENTS FOR NUCLEAR POWER REACTORS" (EPID L-2018-LLA-0095) DATED SEPTEMBER 24, 2019

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