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

September 17, 2019

Mr. William R. Gideon
Site Vice President
Brunswick Steam Electric Plant
Duke Energy Progress, LLC
8470 River Rd., SE (M/C BNP001)
Southport, NC 28461

SUBJECT: BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2 – ISSUANCE OF AMENDMENT NOS. 292 AND 320 TO ADOPT 10 CFR 50.69, "RISK-INFORMED CATEGORIZATION AND TREATMENT OF STRUCTURES, SYSTEMS, AND COMPONENTS FOR NUCLEAR POWER REACTORS" (EPID L-2018-LLA-0008)

Dear Mr. Gideon:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment Nos. 292 and 320 to Renewed Facility Operating License Nos. DPR-71 and DPR-62 for the Brunswick Steam Electric Plant, Units 1 and 2, respectively. These amendments are in response to your license amendment request dated January 10, 2018, as supplemented by letters dated November 2, 2018, February 13, 2019, and April 8, 2019.

The amendments add a new license condition to the Renewed Facility Operating Licenses to allow the implementation of Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.69, "Risk informed categorization and treatment of structures, systems and components [SSCs] for nuclear power reactors." The provisions of 10 CFR 50.69 allow adjustment of the scope of SSCs subject to special treatment requirements (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation) based on a method of categorizing SSCs according to their safety significance.

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,

Dennis J Galvin

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

Docket Nos. 50-325 and 50-324

Enclosures:
1. Amendment No. 292 to DPR-71
2. Amendment No. 320 to DPR-62
3. Safety Evaluation

cc: Listserv

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

DUKE ENERGY PROGRESS, LLC

DOCKET NO. 50-325

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 292
Renewed License No. DPR-71

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

   A. The application for amendment filed by Duke Energy Progress, LLC (the licensee), dated January 10, 2018, as supplemented by letters dated November 2, 2018, February 13, 2019, and April 8, 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 3 of Renewed Facility Operating License No. DPR-71 is hereby amended to read as follows:

   Additional Conditions

   The Additional Conditions contained in Appendix B, as revised through Amendment No. 292, 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

Eddie Miller for US

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 17, 2019

ATTACHMENT TO LICENSE AMENDMENT NO. 292

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 1

RENEWED FACILITY OPERATING LICENSE NO. DPR-71

DOCKET NO. 50-325

Replace the following pages of Renewed Facility Operating License No. DPR-71 and Appendix B, Additional Conditions, with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the area of change.

Renewed Facility Operating License

Remove Page	Insert Page
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Appendix B, Additional Conditions

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3. Additional Conditions

The Additional Conditions contained in Appendix B, as revised through Amendment No. 292, are hereby incorporated into this license. Duke Energy Progress, LLC shall operate the facility in accordance with the Additional Conditions.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

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

Attachments:
1. Unit 1 – Technical Specifications – Appendices A and B

Date of Issuance: June 26, 2006

Amendment Number	Additional Conditions	Implementation Date
292	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 1 License Amendment No. 292 dated September 17, 2019. Duke Energy will complete the implementation items list in Attachment 1 of Duke letter to NRC dated April 8, 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).	Upon implementation of Amendment No. 292.

UNITED STATES
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WASHINGTON, D.C. 20555-0001

DUKE ENERGY PROGRESS, LLC

DOCKET NO. 50-324

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 320
Renewed License No. DPR-62

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

   A. The application for amendment filed by Duke Energy Progress, LLC (the licensee), dated January 10, 2018, as supplemented by letters dated November 2, 2018, February 13, 2019, and April 8, 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 3 of Renewed Facility Operating License No. DPR-62 is hereby amended to read as follows:

   Additional Conditions

   The Additional Conditions contained in Appendix B, as revised through Amendment No. 320, 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 17, 2019

ATTACHMENT TO LICENSE AMENDMENT NO. 320

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 2

RENEWED FACILITY OPERATING LICENSE NO. DPR-62

DOCKET NO. 50-324

Replace the following pages of Renewed Facility Operating License No. DPR-62 and Appendix B, Additional Conditions, with the attached revised pages. The revised pages are identified by amendment number and contains marginal lines indicating the area of change.

Renewed Facility Operating License

Remove Page	Insert Page
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Appendix B, Additional Conditions

Remove Page	Insert Page
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M. Mitigation Strategy License Condition

Develop and maintain strategies for addressing large fires and explosions and that include the following key areas:

(1) Fire fighting response strategy with the following elements:
1. Pre-defined coordinated fire response strategy and guidance
2. Assessment of mutual aid fire fighting assets
3. Designated staging areas for equipment and materials
4. Command and control
5. Training of response personnel

(2) Operations to mitigate fuel damage considering the following:
1. Protection and use of personnel assets
2. Communications
3. Minimizing fire spread
4. Procedures for implementing integrated fire response strategy
5. Identification of readily-available pre-staged equipment
6. Training on integrated fire response strategy
7. Spent fuel pool mitigation measures

(3) Actions to minimize release to include consideration of:
1. Water spray scrubbing
2. Dose to onsite responders

N. The licensee shall implement and maintain all Actions required by Attachment 2 to NRC Order EA-06-137, issued June 20, 2006, except the last action that requires incorporation of the strategies into the site security plan, contingency plan, emergency plan and/or guard training and qualification plan, as appropriate.

3. Additional Conditions

The Additional Conditions contained in Appendix B, as revised through Amendment No. 320, are hereby incorporated into this license. Duke Energy Progress, LLC shall operate the facility in accordance with the Additional Conditions.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

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

Attachments:
1. Unit 2 – Technical Specifications – Appendices A and B

Date of Issuance: June 26, 2006

Amendment Number	Additional Conditions	Implementation Date
320	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. 320 dated September 17, 2019. Duke Energy will complete the implementation items list in Attachment 1 of Duke letter to NRC dated April 8, 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).	Upon implementation of Amendment No. 320.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 292 AND 320

TO RENEWED FACILITY OPERATING LICENSE NOS. DPR-71 AND DPR-62

DUKE ENERGY PROGRESS, LLC

BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2

DOCKET NOS. 50-325 AND 50-324

1.0 INTRODUCTION

By letter dated January 10, 2018 (Reference 1), as supplemented by letters dated November 2, 2018 (Reference 2), February 13, 2019 (Reference 3), and April 8, 2019 (Reference 4), Duke Energy Progress, LLC (Duke Energy, the licensee) submitted a license amendment request (LAR) for the Brunswick Steam Electric Plant, Units 1 and 2 (Brunswick or BSEP). The licensee proposed to add a new license condition to the Renewed Facility Operating Licenses to allow the implementation of Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.69, "Risk informed categorization and treatment of structures, systems and components [SSCs] for nuclear power reactors." The provisions of 10 CFR 50.69 allow adjustment of the scope of SSCs subject to special treatment requirements (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation) based on a method of categorizing SSCs according to their safety significance.

To support its review, the U.S. Nuclear Regulatory Commission (NRC or the Commission) staff conducted a regulatory audit that consisted of (1) a remote audit, (2) an onsite audit at the Duke Energy Corporate office in Charlotte, North Carolina from July 17-19, 2018, and (3) an onsite audit at Brunswick from August 21-23, 2018. The remote audit and the August 2018 onsite audit were focused on the high winds (HW) and external flooding (XF) probabilistic risk assessments (PRAs) while the July 2018 onsite audit focused on the overall categorization process. The NRC staff issued corresponding audit plans on May 8, 2018, July 2, 2018, July 11, 2018, and August 7, 2018 (References 5, 6, 7, and 8, respectively) and audit summaries on November 14, 2018, and July 1, 2019 (References 9 and 10, respectively).

By e-mails dated October 9, 2018 (Reference 11), January 14, 2019 (Reference 12), and March 7, 2019 (Reference 13), the NRC staff transmitted requests for additional information (RAIs) to the licensee. By letters dated November 2, 2018, February 13, 2019, and April 8, 2019, the licensee responded to the requests. The supplements provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on May 22, 2018 (83 FR 23731).

2.0 REGULATORY EVALUATION

2.1 Risk-Informed Categorization and Treatment of SSCs

A risk-informed (RI) approach to regulation enhances and extends the traditional deterministic regulation by considering risk in a comprehensive manner.

Specifically, an RI 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. 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 using 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, alternative treatment requirements may be implemented in accordance with the regulation. For SSCs determined to be of high safety significance, the requirements set forth in 10 CFR 50.69(b)(1)(i) through 50.69(b)(1)(xi), and 50.69(g) shall apply.

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

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

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 FR 68008, 68011 (November 2, 2004).

2.2 Licensee's Proposed Changes

The licensee proposed in the LAR to amend its Renewed Facility Operating Licenses. On October 9, 2018, the NRC staff requested further clarification of the license change in PRA-RAI-09 (Reference 11). The licensee responded (Reference 2) by adding and updating (References 3 and 4) the following license condition that would allow for the implementation of 10 CFR 50.69:

> 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 [American Society of Mechanical Engineers]/ANS [American Nuclear Society] PRA Standard RA-Sa-2009; as specified in Unit 1 [Unit 2] License Amendment No. [XXX] dated [DATE].
>
> Duke Energy will complete the implementation items list in Attachment 1 of Duke letter to NRC dated April 8, 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 [Regulatory Guide] 1.200, Revision 2), and any findings will be resolved and reflected in the PRA of record prior to implementation of the 10 CFR 50.69 categorization process.
>
> Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from a seismic margins approach to a seismic probabilistic risk assessment approach).

2.3 Regulatory Review

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

Regulatory Requirements

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

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

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

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

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

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

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

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,

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

and must be subjected to a peer review process assessed against a standard or set of acceptance criteria that is endorsed by the NRC.

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

(iii) Maintain defense-in-depth [(DID)].

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

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

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

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

Guidance

Nuclear Energy Institute (NEI) 00-04, Revision 0, "10 CFR 50.69 SSC Categorization Guideline," July 2005 (Reference 14), describes a process for determining the safety significance of SSCs and categorizing them into the four RISC categories defined in 10 CFR 50.69. This categorization process is an integrated decision-making process that incorporates risk and traditional engineering insights. 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. As stated in 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," May 2006 (Reference 15), such non-PRA-type evaluations will result in more conservative categorization, in that special treatment requirements will not be allowed to be relaxed for SSCs that are relied upon in such evaluations that are categorized as HSS. The degree of relaxation that the NRC will accept under 10 CFR 50.69 (i.e., SSCs subject to relaxation of special treatment requirements) will be commensurate with the assurance provided by the evaluation.

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

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

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

RG 1.201, Revision 1, 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 seismic margins analysis to a seismic PRA), the licensee or applicant will need to seek NRC approval, via a license amendment, of the implementation of the new approach in their categorization process. In addition, RG 1.201 states that all aspects of NEI 00-04 must be followed.

RG 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," March 2009 (Reference 16), 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 RA-Sa-2009, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," February 2009 ("ASME/ANS 2009 Standard" or "PRA Standard") (Reference 17). This RG provides guidance for determining the technical acceptability 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 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," January 2018 (Reference 18), 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.

The NRC staff performed regulatory audits to confirm details of the development of HW and XF PRAs. The staff reviewed licensee documents, performed a walkdown of the plant, and discussed the staff's concerns with the licensee during the audits.

3.0 TECHNICAL EVALUATION

3.1 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 states that a licensee voluntarily choosing to implement 10 CFR 50.69 shall submit an application for license amendment under 10 CFR 50.90 that contains a description of the process for categorization of RISC-1, RISC-2, RISC-3, and RISC-4 SSCs. 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 provides that the categorization process described in NEI 00-04, with any noted exceptions or clarifications, is acceptable for implementation of 10 CFR 50.69. 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. DID 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. Integrated Development Plan Review and Approval (Section 9 of NEI 00-04); and
8. SSC Categorization (Section 10 of NEI 00-04).

The licensee stated in the LAR that it will implement the risk categorization process in accordance with NEI 00-04, as endorsed by RG 1.201. The LAR provided details of the categorization process as follows: (1) summary 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 3.3 of this SE).
- Defining system function and assigning components to functions (see Section 3.4 of this SE).
- Risk Characterization. Safety significance of active components is assessed through a combination of PRA and non-PRA methods, covering all hazards (see Section 3.5 of this SE).
- DID characterization performed in accordance with Section 6 of NEI 00-04 (see Section 3.6 of this SE).
- Passive Characterization. Passive components are not modeled in the PRA, and therefore, a different assessment method is used to assess the safety significance of these components. 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 (see Section 3.5.4 of this SE).
- Qualitative Characterization. System functions are qualitatively categorized as HSS or LSS based on the seven questions in Section 9.2 of NEI 00-04 (see Section 3.9 of this SE). The licensee confirmed in the response to PRA-RAI-02 (Reference 2) 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 of the LAR (Table 1 below), 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 LAR Section 3.1.1, the licensee explained that consistent with NEI 00-04, the categorization of a component or function is "preliminary" until it has been confirmed by the IDP. The licensee

stated that a component or function is preliminarily categorized as HSS if any element of the process results in a preliminary HSS determination. This preliminary categorization will be presented to the IDP for review. The IDP will decide the final categorization as discussed in Section 3.9 of this SE.

In LAR Section 3.1.1, the licensee provided clarifications 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), some steps are performed at the function level (e.g., qualitative criteria), and some steps are performed at the function and component level (e.g., shutdown, core damage DID).

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

As discussed in Section 3.9 of this SE, the licensee explained in LAR Section 3.1.1 that the seven qualitative criteria are assessed preliminary by the 50.69 categorization team, prior to the IDP. The licensee further clarified that if the IDP determines that any one of the seven qualitative criteria cannot be confirmed (false response) for a system function, then the final categorization of that function will be HSS.

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, as endorsed by RG 1.201.

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

The licensee's risk categorization process uses PRAs to assess risks from internal events (including internal flooding), internal fire, HW, and XF. For the other applicable risk hazard groups, the licensee's process uses non-PRA methods for the risk characterization. The licensee uses its SMA to assess seismic risk, and its shutdown safety plan to assess shutdown risk. The use of risk information and quality of PRA is reviewed in Section 3.5 of this SE.

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

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

Section 4 of NEI 00-04 states that system selection and boundary definition include defining system boundaries where the system interfaces with other systems. Identification of system functions includes identification of all system functions including design-basis and beyond design-basis functions identified in the PRA and making sure that system functions are consistent with the functions defined in design-basis documentation and maintenance rule functions. The coarse mapping of components to functions involves the initial breakdown of system components into system functions they support. The licensee should then identify, and document system components and equipment associated with each function. However, there may be circumstances where the categorization of a candidate LSS SSC within the scope of the system being considered cannot be completed because it also supports an interfacing system. In this case, the SSC will remain uncategorized until the interfacing system is considered.

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

Section 2.2 of the LAR states that the safety functions in the categorization process include the design-basis functions, as well as functions credited for severe accidents (including external

events). Section 3.1.1 of the LAR summarizes the different hazards and plant states for which functional and risk-significant information will be collected. In addition, Section 3.1.1 of the LAR states that the SSC categorization process documentation will include, among other items, system functions identified and categorized with the associated bases and mapping of components to support function(s).

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

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

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 component safety significance assessment assesses 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 (including internal flooding)
- 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 (HW, 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).

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

- SMA safe shutdown equipment list (SSEL) as a screening process results
- IPEEE screening to assess the risk from other external hazards

- Shutdown safety plan to assess shutdown risk

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

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

The licensee's PRA is comprised of (1) an internal events PRA that calculates CDF and LERF from internal events, including internal flooding, at full power, (2) a fire PRA, (3) an HW PRA, and (4) an XF 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.

3.5.1.1 Internal Events PRA

The NRC staff review of the internal events PRAs was based on the results of the peer review of the internal events PRA; associated facts and observations (F&Os) closure review described in LAR Section 3.3; and previously docketed information on PRA quality submitted to the NRC for relocation of surveillance frequencies to licensee control (Technical Specifications Task Force (TSTF) Traveler TSTF-425), as described in the NRC letter dated May 24, 2017 (Reference 19), and the request to adopt National Fire Protection Agency Standard 805, as described in the NRC letter dated January 28, 2015 (Reference 20). The internal events model was subject to a self-assessment and a full scope peer review in June 2010.

In the LAR, the licensee 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 21) to the guidance in NEI 05-04, "Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard," November 2008 (Reference 22); NEI 07-12, "Fire PRA Peer Review Process Guidelines," June 2010 (Reference 23); and NEI 12-13, "External Hazards PRA Peer Review Process Guidelines," August 2012 (Reference 24), concerning the process "Close-Out of Facts and Observations." Appendix X provides guidance for a licensee's use to close F&Os that were generated during a peer review process. The NRC staff accepted, with conditions, a final version of Appendix X to NEI 05-04, 07-12, and 12-13 ("Appendix X") in the NRC letter dated May 3, 2017 (Reference 25).

The licensee submitted a list of all the open F&Os from the self-assessments and peer reviews including 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. The NRC staff requested a more

detailed discussion that the dispositions will have a minimal impact (both individually and aggregate) on the 10 CFR 50.69 categorization process. The NRC staff requested additional information to clarify the licensee's disposition for some of the findings as described in the following paragraphs.

For F&O 1-19, the NRC staff noted the exclusion of updated component failure data could affect risk categorizations. In its November 2, 2018, response to PRA-RAI-01.a (Reference 2), the licensee explained that the four items of concern are categorized as HSS without the data update and would not affect the categorization process.

For F&O 3-6, the NRC staff noted the issue related to operator action OPER-DCDG was not addressed in the disposition, which could affect risk categorizations. In its November 2, 2018, response to PRA-RAI-01.b (Reference 2), the licensee explained that the plant no longer relies on the Severe Accident Mitigation Alternatives (SAMA) diesel generator (DG) to supply power to the battery chargers and, therefore, this operator action was removed from the PRA.

For F&O QU-C2-1, the NRC staff noted that the peer reviewers commented on the use of joint human error probabilities (JHEPs) floor values (i.e., minimum nominal values) although it was unclear whether floor values were or were not applied. In its November 2, 2018, response to PRA-RAI-01.d (Reference 2), the licensee explained that there were no JHEPs in the internal events that were less than the floor value of 1E-06 in Table 4-3 of Electric Power Research Institute (EPRI) Technical Report (TR) 1021081, "Establishing Minimum Acceptable Values for Probabilities of Human Failure Events," October 2010 (Reference 26). The licensee further clarified that some (currently ten) of the Fire PRA JHEP values were less than the floor value of 1E-05 in Table 2-1 of NUREG-1792, "Good Practices for Implementing Human Reliability Analysis (HRA)," April 2005 (Reference 27), but that each of these JHEP values was justified and the justification documented. The NRC staff finds that JHEP values have been developed consistent with the accepted guidance and, therefore, are acceptable.

The NRC staff discovered during the July 2018 onsite audit (Reference 8) that the Brunswick models incorporated diverse and flexible coping strategies (FLEX) components. The staff's review of the approach for crediting FLEX equipment in PRA models and consistency with the NRC memorandum dated May 30, 2017, "Assessment of the Nuclear Energy Institute 16-06, 'Crediting Mitigating Strategies in Risk-Informed Decision Making'" (Reference 28) is discussed in the paragraphs below.

In its November 2, 2018, response to PRA-RAI-08.a (Reference 2), the licensee stated that only the internal events PRA model incorporates the FLEX DGs, FLEX portable pumps powered either by their own diesel engine or by an NRC Order B.5.b response diesel pump, and FLEX air compressors.

In its November 2, 2018, response to PRA-RAI-08.b (Reference 2), the licensee stated that the failure rates for FLEX equipment were based on the generic industry data provided in NUREG/CR-6928, "Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants," February 2007 (Reference 29), developed for similar non-FLEX equipment. Generic data was used since no plant-specific data is currently available and there is no industry data for FLEX equipment. The licensee also noted that the nonsafety-related FLEX DGs used the generic data for the safety-related emergency DGs (EDGs).

In its February 13, 2019, response to PRA-RAI-08.01.a (Reference 3), the licensee stated that plant-specific data on FLEX DGs has been compiled across the Duke Energy fleet (Brunswick, Robinson, Harris, McGuire, Oconee, and Catawba). The licensee stated that the current data set contains the results from over 200 tests of varying scope (i.e., full load, 50-percent load, 20-percent load, other load, no load) and frequency (i.e., monthly, quarterly, yearly, biennial, triennial) and that, for the FLEX DGs, the plant-specific failure rate has been determined in accordance with the PRA standard. The licensee noted that the start failure rate of the Brunswick FLEX DGs is between that of generic EDGs and station blackout (SBO) DGs per the 2015 standardized plant analysis risk component unreliability data. However, because of the relatively short run times during testing, the licensee did not consider the current data results sufficient for computing a plant-specific run failure rate. As a result, the licensee will use the standardized plant analysis risk SBO EDG run failure rate as the plant-specific run failure rate.

The licensee performed a sensitivity study to assess the impact of the FLEX DG failure rates on categorization results. The licensee calculated the internal events CDF using the plant-specific data. Next, the licensee increased the failure data a factor of 3 and recalculated the CDF. The increase in CDF was less than 2E-10 for each unit. The licensee repeated the process for LERF; there was no increase for either unit. Based on the negligible increase in CDF and no increase in LERF, the NRC staff finds that changes in the failure rate values for the FLEX DGs should have a negligible impact on the importance measures used in categorization.

Regarding the HRA methodology used for FLEX operator actions, the licensee stated in its November 2, 2018, response to PRA-RAI-08.c (Reference 2) that the FLEX human failure events (HFEs) were evaluated in accordance with ASME/ANS PRA Standard Supporting Requirement (SR) HR-G3. For pre-initiator human failures, the licensee did not perform an evaluation to determine HFE failure probability but used the screening values based on NUREG/CR-4772, "Accident Sequence Evaluation Program Human Reliability Analysis Procedure," February 1987 (Reference 30). The NRC staff finds the use of NUREG/CR-4772 screening values for pre-initiator failures adequate since the value incorporates both a commission and omission element. In the NRC's May 30, 2017, memorandum (Reference 28), the NRC staff, for crediting FLEX equipment in PRAs, stated that until gaps in the HRA methodologies are addressed by improved industry guidance, human error probabilities (HEPs) associated with actions for which the existing approaches are not explicitly applicable should be submitted to the NRC for review. Given that ASME/ANS SR HR-G3 is part of the gap in the methodology and it appeared the licensee made no further adjustments to address this gap, in PRA-RAI-08.01.b transmitted January 14, 2019 (Reference 12), the NRC requested the licensee to provide details on the FLEX HFE development and to describe how the HRA methodology gap issue was addressed.

In its February 13, 2019, response to PRA-RAI-08.01.b (Reference 3), the licensee stated that as part of the 50.69 categorization process, NEI 00-04 requires sensitivity studies for each system categorized to "Decrease all human error basic events to their 5th percentile value and increase all human error basic events to their 95th percentile value." The licensee stated that the FLEX operator HEPs will be included in these sensitivity studies to determine the impact that these FLEX operator actions have on equipment importance and that results of the sensitivities will be provided to the IDP in accordance with NEI 00-04. In addition, the licensee stated that Duke Energy will continue to stay informed of ongoing industry initiatives associated with modeling of FLEX operator actions and as new methodologies become available for industry use, they will be reviewed and implemented in accordance with Duke Energy's PRA model update process.

Given the inclusion of FLEX equipment in the PRA model occurred after the 2010 peer review, NRC requested a determination if the model addition of FLEX constituted a PRA Upgrade as defined by the ASME/ANS PRA Standard. In its November 2, 2018, response to PRA-RAI-08.d (Reference 2), the licensee stated that no new methodologies were used to incorporate FLEX in the PRA model, but noted that the SBO sequence moved, "from the risk significant category to the non-risk significant category." The NRC staff notes that the second criterion for PRA upgrade is "change in scope that impacts the significant accident sequences or the significant accident progression sequences." In its January 14, 2019, RAI (Reference 12), the NRC staff questioned how the addition of FLEX equipment in the Brunswick internal events model was not a PRA upgrade.

In its February 13, 2019, response to PRA-RAI-08.01.c (Reference 3), the licensee clarified that Brunswick installed small DGs (called SAMA diesels). The SAMA diesels' primary risk-significant function was to charge the batteries during an SBO. The SAMA diesels were incorporated into the PRA model prior to the last full-scope internal events peer review in June 2010. Inclusion of the SAMA diesels was within the scope of that internal events peer review. In the last internal events model update, which occurred in 2017, the function for charging the batteries during an SBO was shifted from the SAMA diesels to the FLEX DGs to reflect modifications completed at Brunswick. The licensee stated that the FLEX DGs have been modeled in the PRA using the same methods that were previously utilized for the SAMA diesels. The NRC staff finds that changing that function to the FLEX DG does not constitute a significant change in scope or capability of the model and, therefore, no PRA Upgrade has been implemented and a peer review is not required.

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. Paragraph 50.69(b)(2)(iii) requires that a licensee submit the results of the PRA review process conducted to meet § 50.69(c)(1)(i). RG 1.200 provides guidance for determining the technical adequacy of internal events PRA by comparing the PRAs 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). Therefore, the NRC staff concludes that the quality of the internal events PRA with the completion of the implementation item (ii) as part of the proposed license condition meets the requirements in 10 CFR 50.69(c)(1)(i).

3.5.1.2 Internal Flooding PRA

The internal flooding model was subject to a self-assessment and a full-scope peer review in June 2010 followed by a focused-scope peer review in December 2016. In August 2017, an F&O closure review was performed by an independent team on all internal events finding level F&Os.

The licensee submitted a list of all the open F&Os from the self-assessments and peer reviews including 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. The NRC staff requested a more detailed discussion demonstrating that the dispositions will have a minimal impact (both

individually and aggregate) on the 10 CFR 50.69 categorization process. The NRC staff requested additional information to clarify the licensee's disposition for some of the findings as described in the following paragraphs.

For F&O IFSN-A8, the NRC staff noted the exclusion of expansion joint failures could affect risk categorizations. In its November 2, 2018, response to PRA-RAI-01.c (Reference 2), the licensee explained that the expansion joints are in pits in the Turbine Building and if an expansion joint were to rupture, the level in the corresponding pit would rapidly increase. The licensee stated that level sensors in these pits would then trip the circulating water intake pumps, which would cause a plant trip. Floor drains within these pits are sealed and plant design features would prevent gravity flow of water into the Turbine Building once the plant and the pumps trip. As a result, the rupture of the expansion joints would be a self-extinguishing event that is contained within the condenser pits. The licensee stated that as there is no additional consequence from the expansion joint rupture, this event is already represented by the internal events model and additional consideration in the internal flooding PRA is not needed. In addition, the licensee evaluated the probability of flooding from the expansion joint with the subsequent failure of the level switches, or operators to trip the circulating water pumps. The licensee noted that the flood scenario can be screened using the ASME/ANS PRA Standard SR IE-C6.

For F&O IFEV-A5, the NRC staff noted issues with the method of calculating flood-initiating event frequencies that could affect risk categorizations, specifically that the licensee used potentially incorrect pipe break frequency calculations. In its November 2, 2018, response to PRA-RAI-01.e (Reference 2), the licensee explained that the pipe break frequencies were calculated correctly. During the July 2018 onsite audit (Reference 9), the NRC staff reviewed the independent assessment team (IAT) requirements to close out the finding. The IAT determined that the calculation method was appropriate, and the finding remained open until the Unit 1 internal flooding PRA model was updated with these calculations. The NRC audit team was informed by the licensee that the Unit 1 internal flooding model, to be used for categorization, has been updated with the new pipe break frequencies.

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. Paragraph 50.69(b)(2)(iii) requires that a licensee submit the results of the PRA review process conducted to meet § 50.69(c)(1)(i). RG 1.200 provides guidance for determining the technical adequacy of internal flooding PRA by comparing the PRAs 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). Therefore, the NRC staff concludes that the quality of the internal flooding PRA, meets the requirement in 10 CFR 50.69(c)(1)(i).

3.5.1.3 Fire PRA

The NRC staff reviewed the results of the peer review of the fire PRA and associated F&O closure review described in LAR Section 3.3 and presented in LAR Attachment 3. The licensee's fire PRA was subject to a self-assessment and full-scope industry peer review in February 2012 followed by a focused-scope peer review in May 2015. In August 2017, an F&Os closure review was performed by an independent team on fire events finding level F&Os.

The licensee submitted a list of all the F&Os from the self-assessments and peer reviews including the F&Os that were resolved by 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. The NRC staff requested a more detailed discussion demonstrating that the dispositions will have a minimal impact (both individually and aggregate) on the 10 CFR 50.69 categorization process. The NRC staff requested additional information to clarify the licensee's disposition for some of the findings as described in the following paragraphs.

For F&O 1-34, the NRC staff noted the exclusion of updated fire barrier failure probabilities could affect risk categorizations. In its November 2, 2018, response to PRA-RAI-01.f (Reference 2), the licensee explained that the appropriate fire barrier failure probabilities have been applied with an associated fire PRA model update. The F&O resolution was subsequently reviewed and closed (i.e., found acceptable) by an October 2018 industry IAT F&O closure review that was performed to Appendix X.

For F&O 4-1, the NRC staff noted the exclusion of the appropriate fire severity factors could affect risk categorizations. In its November 2, 2018, response to PRA-RAI-01.g (Reference 2), the licensee explained the cabinet breaching factor was updated with the appropriate fire severity factor. The F&O resolution was subsequently reviewed and closed (i.e., found acceptable) by an October 2018 industry IAT F&O closure review that was performed to Appendix X.

For F&O 6-4, the NRC staff noted the description and disposition related to this finding pertaining to fire barrier failure for multi-compartment analysis did not appear complete and requested clarification. In its November 2, 2018, response to PRA-RAI-01.h (Reference 2), the licensee explained that the finding that the fire barrier failure rates used in the Brunswick fire model are those prescribed in NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities," Volumes 1 and 2, September 2005 (References 31 and 32, respectively); however, the worst-case value for failure probability of the barrier was used. To correct this issue, the licensee applied the summed fire barrier failure probabilities, rather than the worst case (as was previously in the model). The F&O resolution was subsequently reviewed and closed (i.e., found acceptable) by an October 2018 industry IAT F&O closure review that was performed to Appendix X.

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. Paragraph 50.69(b)(2)(iii) requires that a licensee submit the results of the PRA review process conducted to meet § 50.69(c)(1)(I). RG 1.200 provides guidance for determining the technical adequacy of a fire PRA by comparing the PRA to the relevant parts of the ASME/ANS 2009 Standard using a peer review process. Based on its review, the NRC staff finds that the licensee has followed the guidance in RG 1.200 and submitted the results of the peer review, and therefore, meets the requirement in 10 CFR 50.69(b)(2)(iii). Therefore, the NRC staff concludes that the quality of the fire PRA meets the requirement in 10 CFR 50.69(c)(1)(i).

3.5.1.4 High Winds and External Flooding PRAs

In Section 3.2.4 of the LAR, the licensee stated that the proposed categorization process will use peer-reviewed HW and XF PRA models. The NRC staff's review of the technical acceptability of the HW and XF PRA models for this application is discussed below. The NRC staff performed regulatory audits as part of its review of the HW and XF PRAs (References 5, 8, and 10) during which the staff reviewed licensee documents, performed a walkdown of the plant, and discussed the staff's concerns with the licensee.

3.5.1.4.1 Evaluation of External Hazards PRA Peer Review Process

In Section 3.3 of the LAR, the licensee stated that the Brunswick HW and XF PRA models were subject to a full-scope peer review in February 2012 in accordance with the guidance in RG 1.200, Revision 2. Appendices B, C, and D to RG 1.200, Revision 2, provide the NRC regulatory position on the peer review process in NEI 00-02, "Probabilistic Risk Assessment Peer Review Process Guidance," May 2006 (Reference 33), 05-04, and 07-12. Regulatory Position 2.2 of RG 1.200, Revision 2, states that "[a]n acceptable peer review approach is one that is performed according to an established process..." and the peer reviewers' "technical expertise includes experience in performing (not just reviewing) the work in the element assigned for review." Regulatory Position 2.2 in RG 1.200, Revision 2, further states that when "the staff's regulatory positions contained in the appendices are taken into account, use of a peer review can be used to demonstrate that the PRA [with regard to an at-power Level 1/LERF PRA for internal events (excluding external hazards)] is adequate to support a risk informed application." Therefore, RG 1.200, Revision 2, does not endorse any peer review guidance for external hazards. As a result, the NRC staff questioned the process followed by the peer reviewers for performing the peer review of the HW and XF PRAs. NEI 12-13 provides guidance for conducting and documenting peer reviews of external hazard PRAs. The NRC staff accepted the use of NEI 12-13 for performing the peer review of an external hazards PRA for use in risk-informed licensing applications, as modified by the NRC staff's comments and qualifications, in a March 7, 2018, letter (Reference 34). The NRC staff also requested information on the consistency of the process followed for the peer review of the HW and XF PRAs with that in NEI 12-13 as well as justification for any deviations.

In the enclosure to the November 2, 2018, supplement, the licensee stated that NEI 12-13 was issued after the peer review for the licensee's HW and XF PRAs was completed and that the peer review process for the HW and XF PRAs was adapted from the internal events PRA peer review process outlined in NEI 05-04. The licensee provided discussions on the considerations for the duration of the peer review, onsite walkdowns by the peer review team, and the peer review team composition. The licensee stated that the duration of the peer review as well as the size of the peer review team was determined based on scope of the review. The licensee stated that the peer reviewers performed work ahead of the onsite review portion to assess the various SRs for their applicable technical areas, assessed the completeness as well as quality of the PRAs against the SRs and documented their basis for the assessment of each SR as well as high level requirements. The licensee explained that the assessment against each SR was performed using the guidance in RG 1.200 and noted that the language from RG 1.200 was quoted in the peer review report as the basis for capability category (CC) assignments.

On the topic of walkdowns by the peer review team, the licensee cited Section 8-3.3.4 and Section 7-3.3.4 of the 2009 ASME/ANS PRA Standard for the HW and XF PRAs, respectively, which state that the peer review team shall review the plant walkdown but do not require that the peer review team perform a walkdown. The licensee also noted that the NRC staff did not

add qualifications or clarifications to those portions of the 2009 ASME/ANS PRA Standard in Tables A-7 and A-8 of RG 1.200, Revision 2. The licensee explained that the HW and XF peer review team reviewed the walkdown data for the hazards and included their assessment of the walkdown data as part of relevant SRs, which the peer reviewers found to be met at CC-II/III with no findings. On the topic of qualifications of the peer review team, the licensee explained that, consistent with the guidance in Section 1-6.2.2 of the 2009 ASME/ANS PRA Standard, the review team members were knowledgeable of the requirements of the Standard for their area of review and experienced in performing the activities related to their assignments. The licensee stated that all members of the peer review team were experienced PRA personnel and that the technical element lead reviewers were assigned to elements according to their level of knowledge and experience. The licensee also stated that the composition of the peer review team met the intent of ensuring that there was more than one peer reviewer with experience in each technical element under review. The licensee provided a synopsis of the experience in performing the hazard development, fragility, and plant response analysis for each peer reviewer that performed the peer review for the HW and XF PRAs.

In the enclosure to the November 2, 2018, supplement, the licensee also provided a discussion of the consistency of the process followed for the peer review of the HW and XF PRAs with that in NEI 12-13. In addition, the licensee stated that the peer review team for its HW and XF PRAs did not assess the maintenance and update process that was a deviation from the guidance in NEI 12-13. The licensee explained that both the HW and XF PRAs were based on the internal events model and the internal events PRA peer review did assess all maintenance and update related SRs. NEI 12-13 suggests that the peer review team include two utility personnel knowledgeable in the hazard(s) being assessed to facilitate the exchange of ideas and techniques for effective use of external hazard PRA methodologies. The licensee explained that at the time of the peer review of the HW and XF PRAs, there was a scarcity of utility personnel with specialized experience in the hazards being reviewed. The licensee considered the experiences as well as expertise of the peer reviewers to be sufficient to meet the intent of multiple utility participants on the review team.

Based on the NRC staff's review of the submittal and the information in the enclosure to the November 2, 2018, supplement, the NRC staff finds that:

- The process followed by the peer reviewers during the peer review of the licensee HW and XF PRAs is acceptable for this application because it is consistent with the guidance in RG 1.200 and NEI 12-13 with exceptions discussed below in Sections 3.5.1.4.2, 3.5.1.4.3, and 3.5.1.4.4.

- The deviation from NEI 12-13 related to the inclusion of utility participants on the review team does not impact the results of the peer review in the context of this application.

- The deviation from NEI 12-13 related to the review of the maintenance and update process does not impact the results of the peer review in the context of this application because it is addressed by (1) the internal events PRA peer review, which included all maintenance and update related SRs, and (2) the NRC staff's questions on the maintenance process for the HW and XF PRAs discussed in Section 3.10 of this SE.

The NRC staff further evaluated the impact of the lack of an onsite walkdown by the peer review team and the concern about the expertise of the peer reviewers in the technical element of hazard development, especially for the XF hazard. The NRC staff reached its conclusions on these topics via its own onsite walkdown, specific RAIs related to the hazard development, and

an implementation item proposed by the licensee, as discussed in Sections 3.5.1.4.2 through 3.5.1.4.4.

The NRC staff notes that in the response to RAI PRA-RAI-10.a in the enclosure to the November 2, 2018, supplement (page 50 of 108 of the enclosure), the licensee stated that the review of the HW and XF PRAs constituted "only two elements" within the context of PRA technical elements. As defined in Sections 1.2.7 and 1.2.8 of RG 1.200, Revision 2, hazard analysis, fragility analysis, and plant response analysis are the technical elements in each of the HW and XF PRAs. Further, Parts 7 and 8 of the 2009 ASME/ANS PRA Standard endorsed in RG 1.200, Revision 2, which are related to the HW and XF PRAs, respectively, mention three technical elements for the HW and XF PRAs, namely the hazard, the fragility, and the plant response analyses. Therefore, the guidance in RG 1.200, Revision 2, defines technical elements of the HW and XF PRAs as the elements that constitute each PRA and not the entirety of the HW and XF PRA models.

3.5.1.4.2 Onsite Walkdown

As part of the August 2018 onsite audit (Reference 10), the NRC staff conducted an onsite walkdown. The purpose of the walkdown was to ensure the adequacy and acceptability of key information collected by the licensee during its walkdown and subsequently used in the development of the HW and XF PRA models.

The NEI 12-13 guidance states that an important element of each external hazard PRA peer review is the walkdown of the areas of the plant that are deemed to be important by the peer reviewers based on the specific results of that external hazards PRA and based on their expertise. The NRC staff's review identified dominant SSC contributors to the HW and XF PRAs, dominant operator actions that were credited, and topographical features at the site that were considered in the XF PRA. The audit plan for the August 2018 onsite audit (Reference 8) included the walkdown item list.

The NRC staff's walkdown confirmed the following:

- Identification of equipment vulnerable to external hazards (i.e., HW and XF) and appropriate inclusion of corresponding failure modes.
- Location as well as access paths for dominant operator actions and appropriate consideration of environmental factors on those actions.
- Inclusion and credit for plant design features, such as concrete floor and walls, flood barriers, and watertight doors.
- Consideration and inclusion of spatial relationships between SSCs in the context of the HW and XF hazards.

NRC staff observations from the walkdown were included in the RAIs on the XF hazard and plant response.

3.5.1.4.3 External Flood Hazard Development

External flooding mechanisms may lead to flooding in excess of plant grade such that water impinges upon SSCs and challenges plant safety. Therefore, the NRC staff requested description and justification of the approach used to screen out any flooding mechanism.

In the enclosure to the November 2, 2018, supplement (Reference 2), the licensee stated that a flooding mechanism was screened out from the XF PRA if (1) the flooding mechanism was not applicable to the site, or (2) a demonstrably conservative deterministic analysis of the flooding mechanism showed that the elevation of the corresponding flood did not reach the nominal site grade elevation of the buildings that host the SSCs relied upon to bring the plant to safe and stable conditions. The licensee further explained that the analysis used for screening out a flood mechanism was considered to be demonstrably conservative if it (1) accounted for the characteristics of the underlying flooding mechanism to ensure that there was no potential for the flood mechanism to impinge on credited SSCs, (2) relied on the site grade elevation, which is an inherently rugged feature of site topography, is permanent, passive, and not significantly affected by the flooding mechanism, and (3) provided margin between the flood elevation determined from the analysis and the occurrence of cliff-edge effects. The licensee explained that they used demonstrably conservative analyses to determine the flood elevations for flooding from streams and rivers, failure of dams and onsite water control and/or storage structures, and tsunami, and the analyses established, with sufficient margin, that the resulting elevations did not reach the nominal site grade elevation. As a result, the licensee screened out those hazards from consideration in the XF PRA. The licensee also noted that extreme lake flooding was screened out because the site was not located near a lake.

In the enclosure to the November 2, 2018, supplement (Reference 2), the licensee stated that the LIP event would produce standing water above only two door thresholds in the Reactor Building. The licensee, citing the focused evaluation performed for the flood hazard reevaluation in response to the Near-Term Task Force (NTTF) recommendation 2.3, explained that the leakage through those doors would be very minimal and floor drains or stairwells were present to route the water into the Reactor Building basement sump area with ample volume to store the water prior to reaching any SSCs. The licensee explained that sufficient available physical margin existed from the in-leakage collecting in the basement to the lowest SSC at that elevation. The licensee stated that sump pumps were not credited in actively removing water and that based on the actual elevations of the two-door thresholds in the Reactor Building, the actual in-leakage of flood water was anticipated to be lower than that assumed. As a result, the licensee screened out LIP from inclusion in the XF PRA. Therefore, the only external flooding mechanism that was explicitly considered in the XF PRA was storm surge.

The NRC staff reviewed the licensee's screening of the external flooding hazard phenomena. The NRC staff finds that the screening of external flooding hazard phenomena is acceptable for this application because:

1. The flood elevations do not reach the nominal site grade elevation, with sufficient margin, for flooding from streams and rivers, failure of dams and onsite water control and/or storage structures, and tsunamis.

2. Extreme lake flooding is not applicable to the site due to its location.

3. LIP does not pose a flooding concern for all but two doors at the site. Sufficient margin exists between the in-leakage from those doors during LIP and the lowest

SSC without consideration of active features such as sump pumps. Therefore, LIP does not pose a challenge to plant response and consequently, does not impact this application.

The NRC staff's findings on the acceptability of the screening of the external flooding hazard phenomena for this application does not imply generic acceptability of the screening for other RI applications by the licensee.

In the enclosure to the November 2, 2018, supplement (Reference 2), the licensee provided details of the approach used to determine the initiating event frequencies for external flooding from storm surge. The licensee discussed the methodology, input parameter selection, data selection, and consideration of uncertainty.

The licensee explained that:

- The storm surge analysis performed to estimate the storm surge and wave induced flooding associated with hurricanes affecting the site utilized a 500,000-year simulation of synthetic hurricanes occurring in the North Atlantic basin. The hurricane model used by the licensee for that purpose was the same one that was used to develop the hurricane wind hazard curves for the licensee's HW PRA.

- Using the simulated tracks, a hurricane wind speed hazard curve for the site was developed, from which data were retained for a subset of storms that produced gust wind speeds of 90 miles per hour or greater at the site. This reduced storm set comprising of 14,217 storms was used in simulations to develop a baseline assessment of coastal flooding. The licensee indicated that default wind field model parameters in the simulation model were used with the 14,217 storms and the coastal flooding results were ranked to identify the top 100 flooding events. The licensee performed a second set of simulations using the selected 100 storms in conjunction with a set of refined wind field parameters. Ten synthetic hurricanes producing the highest storm surge at the site were selected from the second set of storm surge simulations and used for more detailed storm surge modeling.

- A set of storm surge and wind wave simulations for the 10 synthetic hurricanes were conducted using a coupled storm surge and wave model to predict the surge and wave response to the pre-determined hurricane storm tracks with the refined wind field model. To improve the representation of the model area, the computational grid for the model was revised to include more accurate topographic details. Available digital imagery, orthophotographs, and light detection and ranging were used to refine the grid resolution. The result of this analysis was the estimated still water elevation levels, including wave setup and static tidal height, and maximum significant wave heights at 14 locations around the site for each of the 10 synthetic hurricanes.

- Wave runup was not considered in the storm surge analysis due to the flat nature of the site, which would result in negligible effect of wave runup calculations on the overall results of the storm surge analysis.

- Since the analysis determined the highest level of storm surge at the site from the 10 synthetic hurricanes that produce such surges over the 500,000 years, the storm that produces the maximum surge is a single realization of the 500,000-year storm surge

level resulting in an initiating frequency of $2x10^{-6}$ per year and the storm with the 10th highest storm surge is a single realization of the 50,000-year surge level resulting in an initiating frequency of $2x10^{-5}$ per year.

- In the XF PRA supporting this application, a point estimate from the licensee's June 1995 IPEEE (Reference 36) was used, which corresponds to a 20-foot (ft) still water flood event and has an initiating event frequency of $7.4x10^{-4}$ per year. The XF PRA did not develop a family of hazard curves.

The licensee also provided a list of the data sources used to develop the 500,000-year synthetic hurricane storm set used in the analysis.

The NRC staff notes that based on the information presented in the November 2, 2018, supplement (Reference 2), the storm surge hazard development (1) lacks a formal uncertainty analysis, (2) uses point estimate and lacks an approach to determine the mean estimate of the occurrence frequency for the flood elevations, (3) lacks inclusion of wave runup in the analysis, and (4) lacks discussion of comparisons between the results of the simulations performed for screening the storms (i.e., identifying the 10 synthetic hurricanes producing the highest storm surge at the site) and the refined analysis to develop confidence in the screening.

In Attachment 1 of the enclosure to the April 8, 2019, supplement (Reference 4), the licensee proposed, as implementation item (i) as part of the proposed license condition, to complete a focused-scope peer review of the XF PRA model hazard development, as well as to resolve and close any resulting findings, per an NRC-approved process, prior to implementation of the licensee's 10 CFR 50.69 program. The staff's review of the implementation item finds that focused-scope peer reviews are performed against the applicable high-level requirements and SRs in the 2009 version of the ASME/ANS PRA Standard, as endorsed in RG 1.200, Revision 2, which include consideration of the technical details that were lacking in the current storm surge hazard development discussed above. Further, the staff's review finds that peer reviews, including focused-scope peer reviews, are performed by individuals with appropriate qualifications as stated in 2009 ASME/ANS PRA Standard, RG 1.200, Revision 2, and, in case of external hazard PRAs, NEI 12-13. In addition, the proposed implementation item includes closure of any findings resulting from the focused-scope peer review using an NRC-accepted process. Therefore, an additional review by qualified individuals will be performed to determine whether technical concerns raised by the focused-scope peer reviewers in the form of findings are appropriately resolved. Based on the use of the endorsed PRA Standard and NRC-approved process to close findings, the NRC staff concludes that after the completion of implementation item (i) in Attachment 1 of the enclosure to the April 8, 2019, supplement as part of the proposed license condition, the inclusion of the storm surge analysis in the XF PRA will be consistent with the guidance in RG 1.200, Revision 2, and the 2009 ASME/ANS PRA Standard endorsed therein.

3.5.1.4.4 High Winds Hazard Development

In the enclosure to the November 2, 2018, supplement (Reference 2), the licensee provided details on the hazard development approach followed for the HW PRA. The licensee explained that four types of extreme winds that can affect the site were included in the wind hazard, namely hurricanes, thunderstorms, extra-tropical storms, and tornadoes. The licensee stated that individual mean wind hazard curves were developed for each wind hazard type and an overall combined mean wind hazard curve was developed, which was used for the HW PRA.

The licensee provided details about the development of the wind hazard curve for each wind hazard type including information on the source(s) of data, the process used to develop the hazard curves, and consideration of uncertainties in parameter values. The licensee also identified the sources of model uncertainty and key assumptions for the wind hazard development. The licensee explained that those sources were either modeled with uncertainties that were fully propagated as part of the derivation of the mean hazard curve and estimated percentiles or they were modeled as random variables in the model and sampled using statistical sampling techniques. The licensee stated that model uncertainties and key assumptions were not considered for the thunderstorm and extra-tropical storm hazard because the wind hazard analyses showed that these hazard types had a negligible effect on the wind hazard and consequently, the HW PRA.

Based on the information in the November 2, 2018, supplement (Reference 2), the NRC staff finds that the licensee's approach for the development of the wind hazard used in its HW PRA is acceptable for this application because:

1. It included the wind hazard types that can affect the site.

2. It used relevant data from appropriate sources for each wind hazard type.

3. The development of the hazard curves for each wind hazard type was performed using technically sound approaches.

4. Model uncertainties and assumptions for each wind hazard type were identified and appropriately considered in the development of the corresponding hazard curves.

5. An overall combined mean wind hazard curve was developed from mean hazard curves for each wind hazard type for use in the HW PRA.

In addition to SSC failure occurring due to the direct impact of wind, the HW PRA, following the 2009 ASME/ANS PRA Standard, also needs to include the impact of SSC failures induced by wind-generated missiles. The NRC staff requested information about the approach followed for the evaluation and development of wind-generated missile hazard. The NRC staff also requested justification, in the context of this application, for any deviation(s) from using a plant-specific high-wind missile analysis methodology for determining the frequency of damage resulting from missiles generated by HW and tornadoes on individual SSCs.

In the enclosure to the November 2, 2018, supplement (Reference 2), the licensee described the approach for identification of the number, type, and location of potential missiles. The licensee stated that a detailed survey of the site was conducted to develop an inventory of potential wind missiles. The licensee provided additional details on the performance and results of the survey including tabulation of the type and number of missiles in different site 'zones.' The NRC staff finds the licensee's approach for determining the number, type, and location of potential missiles at the site to be acceptable for this application because:

1. It was performed in a systematic manner, within 2,500 feet from SSCs included in the HW PRA model, consistent with the missile inventory surveys performed for probabilistic tornado-generated missile impact applications reviewed by the NRC staff.

2. It included sufficient detail in the identification and classification of missile types in each location of the plant that was surveyed.

3. It included consideration of structural missiles that would be generated based on building deconstruction and missile sources due to the inventories within buildings.

The licensee stated that a simplified method for assessing the probability of missile strikes on SSCs was developed that used plant-specific data inputs into a multivariate model derived from previous TORMIS analyses of other plants. TORMIS is a probabilistic tornado-generated missile impact simulation that employs Monte Carlo techniques in order to propagate the transport of tornado-generated missiles and to assess the probability of missile strikes on SSCs.

The licensee explained that missile hit and damage data from two previous TORMIS analyses of nuclear power plants was used to develop a multivariate statistical model to estimate wind missile hit and damage for the licensee's SSCs. The licensee further explained that the model was developed from a statistical analysis of the TORMIS inputs as well as results and was focused on identifying the site-specific parameters that best explain missile hit and damage probability at any given nuclear power plant. The licensee stated that a regression analysis of the model against the data from the TORMIS analyses demonstrated that about 70 percent of the variance in missile hit probability is explained by the model implemented by the licensee. In addition, the licensee also used a function for determining the conditional probability of perforation of steel-plated targets that was based on data from previous TORMIS analyses. The licensee stated that plant-specific inputs such as missile inventory, orientation, and target inventory were used as inputs to the simplified method. The licensee identified the simplified method as a key assumption.

On October 26, 1983, the NRC staff approved the use of TORMIS for assessing the need for positive tornado missile protection for specific safety-related plant features in accordance with the criteria in SRP Section 3.5.1.4 (Reference 37). The NRC staff's review of the licensee's approach to address this key assumption is discussed below.

3.5.1.4.5 Evaluation of Key Assumptions and Sources of Uncertainty for HW and XF PRAs

Section 3.3 of RG 1.200, Revision 2, identifies two aspects necessary to demonstrate the technical acceptability of the PRA: (1) assurance that the pieces of the PRA used in the application have been performed in a technically correct manner, and (2) assurance that the assumptions and approximations used in developing the PRA are appropriate. Section 3.3.2 of RG 1.200, Revision 2, further discusses the second aspect (2, above) and clarifies "[f]or each application that calls upon this regulatory guide, the applicant identifies the key assumptions and approximations relevant to that application. This will be used to identify sensitivity studies as input to the decision-making associated with the application." RG 1.200, Revision 2, defines the terms "key assumption" and "key source of uncertainty" in Section 3.3.2, "Assessment of Assumptions and Approximations."

Section 3.2.7 of the LAR cited certain references for the process of identifying model uncertainties but did not elaborate on the implementation by the licensee. The same section further stated that key PRA model-specific assumptions and sources of uncertainty for this application have been identified and dispositioned in LAR Attachment. Because the LAR did not discuss the licensee's approach for identifying and dispositioning key assumptions and uncertainties, the NRC staff requested the licensee to describe the approach used to identify and characterize the "key" assumptions and "key" sources of uncertainty in the licensee's HW and XF PRA models as well as the disposition to the identified sources for this application.

In the enclosure to the April 8, 2019, supplement (Reference 4), the licensee provided a discussion of the approach for identification of key assumptions and sources of uncertainty for the HW and XF PRA in the context of this application. The licensee stated that its response in the enclosure superseded its previous responses on this topic in the letter dated November 2, 2018 (Reference 2), except for the response to RAI 17-1 in the letter dated February 13, 2019 (Reference 3). The licensee explained that its approach for identifying key assumptions and sources of uncertainty consisted of:

1. Review of the HW and XF PRA documentation for plant-specific assumptions and uncertainties, and

2. Identification of key assumptions and sources of uncertainty using the corresponding definitions in RG 1.200, Revision 2; NUREG-1855, Revision 1, "Guidelines on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking," March 2017 (Reference 38), or related references. The licensee provided details of the specific considerations used to determine whether an assumption and source of uncertainty was key or not in the enclosure to the April 8, 2019, supplement.

Based on this approach, the licensee identified one key assumption for the XF PRA and three key assumptions for the HW PRA. The licensee proposed dispositions for the identified key assumptions in the context of this application. The key assumptions identified by the licensee and their proposed dispositions for this application are:

1. Flood frequency calculation related to the XF PRA: The licensee stated that if the flood frequency determined from the 500,000-year synthetic storm set is implemented into the Brunswick XF PRA, it would constitute a key assumption and a sensitivity study would be completed for each system categorized that evaluated the impact of flood frequency one order of magnitude higher and lower than the value used in the XF PRA. The results of the sensitivity study would be presented to the IDP. The licensee noted that the current flood frequency in the XF PRA is two orders of magnitude higher than that from the 500,000-year storm set and that the use of 500,000-year storm set flood frequency as well as the corresponding sensitivity was tied to implementation item (i) in Attachment 1 of the April 8, 2019, supplement.

2. Missile fragility related to the HW PRA: The licensee identified the use of the simplified model for missile failure probability in the licensee's HW PRA and its ability to reasonably represent the plant as a key assumption. The licensee stated that the impact of this assumption would be evaluated by a sensitivity study where the probabilities of failure due to missile strike would all be multiplied by 2 and a separate case where the probabilities of failure due to missile strike would be divided by 2. The licensee further stated that these factors for the sensitivity were determined based on engineering judgement from subject matter experts. The licensee explained that the sensitivity study would be completed for each system that is categorized along with those described in Table 5-5 of NEI 00-04 and the results would be presented to the IDP.

3. High-wind-induced loss-of-offsite power (LOOP) related to the HW PRA: The licensee explained that a high-wind-induced LOOP event was modeled in the HW PRA using the switchyard relay house fragility as a surrogate for the occurrence probability for such an event. The licensee identified this modeling approach as a key assumption and the licensee stated that the impact of this assumption would be evaluated by a sensitivity

study where the fragility of the switchyard relay house would be multiplied by 2 (i.e., conditional probability of failure decreases for a given wind speed) and a separate case where the switchyard relay house fragility would be divided by 2 (i.e., conditional probability of failure increases for a given wind speed). The licensee further stated that these factors for the sensitivity study were determined based on engineering judgement from subject matter experts. The licensee explained that the sensitivity study would be completed for each system that is categorized along with those described in Table 5-5 of NEI 00-04 and the results would be presented to the IDP.

4. HRA related to HW PRA: The licensee explained that a multiplier approach was used to perform the HRA for the HW PRA. The licensee stated that the uncertainty associated with HRA development would be addressed by the sensitivity in Table 5-5 of NEI 00-04 that evaluates human error probabilities at their 5th and 95th percentile values for all system categorizations with results presented to the IDP.

The NRC staff reviewed the licensee's discussion of the identification and disposition of key assumptions and sources of uncertainty in the HW and XF PRAs for this application. The NRC staff finds that the licensee's approach, including the criteria used as part of the approach, for identifying the key assumptions and sources of uncertainty is consistent with the guidance in RG 1.200, Revision 2, as well as the discussion in NUREG-1855 and is, therefore, acceptable for this application. The NRC staff's review of the licensee's approach for the identification of key assumptions and sources of uncertainty, and the evaluation of the impact of the identified key assumptions and sources of uncertainty for this application concludes that:

1. the licensee identified appropriate sensitivity studies as input to the decisionmaking associated with the application consistent with the guidance in RG 1.200, Revision 2, and Table 5-5 of NEI 00-04;

2. the factor of 10 change in the flood frequency determined from the synthetic storm set is sufficient to capture the uncertainty in frequency determination subsequent to the completion of implementation item (i) in Attachment 1 of the enclosure to the April 8, 2019, supplement, as part of the proposed license condition;

3. the factor of 2 change to the failure probability due to HW-generated missile impact is sufficient to capture the uncertainty in the simplified model for missile failure probability in the licensee's HW PRA because it is consistent with the mean square error from the regression analysis used to derive the simplified model;

4. the factor of 2 change to the fragility of the switchyard relay house is large enough to sufficiently capture the uncertainty in the occurrence of HW-induced LOOP; and

5. the existing sensitivities in Table 5-5 of NEI 00-04 that evaluate the impact of HEPs at the 5th and 95th percentile values include sufficient variation to capture the uncertainty with using multipliers in the HRA for the HW PRA and a separate sensitivity is not necessary for this uncertainty.

In its November 2, 2018, supplement (Reference 2), the licensee stated that the sensitivity analyses in Table 5-5 of NEI 00-04 as well as those identified as part of the PRA acceptability for HW and XF PRAs will be performed every time SSCs are categorized under 10 CFR 50.69. The licensee, citing the guidance in NEI 00-04, further stated that if it is determined that any changes to the HW and XF PRAs that result in new key assumptions and uncertainties as well

as an evaluation of any new sensitivities would more than minimally affect the categorization results, the risk information and the categorization process would be updated. The licensee also explained that its model update procedures included assessment of the impact of PRA changes on the licensee's RI programs, such as 10 CFR 50.69. The model update procedures also address any new key assumptions and sources of uncertainty as well as an evaluation of any new sensitivities necessary as part of the categorization process. The NRC staff finds that the licensee's PRA maintenance process will address new assumptions and sources of uncertainty as well as corresponding sensitivity studies consistent with the NEI 00-04 guidance.

In summary, the NRC staff finds that (1) the licensee identified and evaluated key assumptions and key sources of uncertainty in its HW and XF PRA consistent with the guidance in Section 3.3.2 of RG 1.200, Revision 2; (2) the licensee identified appropriate sensitivity studies as input to the decisionmaking associated with the application consistent with the guidance in Table 5-5 of NEI 00-04 to identify additional "applicable sensitivity studies" for this application; and (3) the licensee's disposition of the identified key assumptions and key sources of uncertainty is appropriate for this application.

3.5.1.4.6 Evaluation of High Winds and External Flooding Peer Review Findings

According to Sections 7-1.2 and 8-1.2 of the 2009 ASME/ANS PRA Standard, it is assumed that full-scope internal events at-power Level 1, and Level 2 LERF PRAs exist and that those PRAs are used as the basis for the HW and XF PRA. Therefore, the technical acceptability of the internal events PRA model used as the foundation for the HW and XF PRAs is an important consideration. Section 3.3 of the LAR stated that the internal events findings were reviewed and closed using the process documented in Appendix X to NEI 05-04, NEI 07-12, and NEI 12-13, as accepted by NRC in the letter dated May 3, 2017 (Reference 25). However, the LAR did not provide information about the propagation of changes made to the internal events model for closing the finding level F&Os to the HW and XF PRAs. In its November 2, 2018, supplement (Reference 2), the licensee stated that all model changes made to the licensee's internal events PRA to resolve finding level F&Os were reviewed and it was determined that finding level resolutions implemented in the internal events model were captured in the licensee's database in order to ensure implementation into other models as applicable. The licensee provided a brief description of the items in the database that were applicable to the HW and XF PRAs. The items in the database that were applicable to the HW and XF PRAs do not represent major shortcomings in those PRAs and are not expected to significantly impact the categorization using those PRAs. The NRC staff's review finds that (1) the resolutions of finding level F&Os will not significantly impact the use of the HW and XF PRAs for categorization, and (2) the resolutions will be propagated to the HW and XF PRAs because the licensee's process captures such implementation, as applicable.

In its November 2, 2018, supplement (Reference 2), the licensee stated that the assumptions identified in the internal events PRA were reviewed for applicability to the XF PRA and that any assumption in the internal events PRA model that was used in the XF PRA was appropriately modeled. The licensee explained that control rod drive pumps and emergency core cooling system room sump pumps credited for mitigation in the XF PRA were powered from emergency sources. The licensee also explained that the fire water pumps were not credited in the XF PRA and passive components, such as buildings, doors, and penetrations that were not explicitly modeled in the internal events PRA were assessed directly for the XF PRA. The NRC staff finds that the licensee appropriately incorporated assumptions from the internal events PRA in the XF PRA.

In Section 3.3 of the LAR, the licensee stated that the Brunswick HW and XF PRAs were subject to a full-scope peer review in February 2012 against RG 1.200, Revision 2. Section 3.3 of the LAR also stated that findings were subsequently reviewed and closed using the process documented in Appendix X to NEI 05-04, NEI 07-12, and NEI 12-13 as accepted by NRC in the letter dated May 3, 2017 (Reference 25). The licensee cited closure of findings for its internal events, internal flood, HW, and fire PRA models. In the November 2, 2018, supplement (Reference 2), the licensee clarified that two of the four open findings for the XF PRA were reviewed by the closure team while the other two findings were not resolved sufficiently to warrant review by the closure team. The licensee also stated that none of the XF PRA finding level F&Os were closed by the independent assessment team and each of the open findings was dispositioned for the 10 CFR 50.69 application. The NRC staff reviewed the licensee's disposition of all open findings for the XF PRA provided in Attachment 3 of the LAR and considered the potential impact of the findings on this application. The NRC staff requested additional information to clarify the licensee's disposition for the findings as described in the following paragraphs.

Finding XFPR-A11-1, related to SR XFPR-A11, stated that "there is no evaluation of the potential impact of external floods on system recoveries credited in the Level 1 PRA." The resolution discussed staging of personnel and reevaluation of HRA and concluded that "changes made are enough to support" this application. However, the discussion of the resolution did not include sufficient detail to support the NRC staff's review of the impacts of environmental conditions on the staged and unstaged operator actions, and the failures of flood protection features that could prevent operators from performing their actions. Furthermore, finding XFPR-A3-1, related to SRs XFPR-A3, -A5, -A8, and -A10, stated that assurance was needed that external flood-caused failures were modeled and that a systematic review of potential impacts of external flooding was performed. The resolution discussed documentation changes but did not provide information on the systematic review of the potential impacts of external flooding.

In the November 2, 2018, supplement (Reference 2), the licensee stated that an evaluation of the impact of an external flood was performed and based on walkdown observations that identified vulnerabilities associated with executing an operator action due to a 20-ft or a 23-ft still water level flood. The licensee explained that the site's severe weather procedures were also considered in the evaluation and that the procedures recommended pre-staging multiple operators at different locations, if conditions warrant, prior to hurricane arrival. The licensee stated that following the evaluation (1) credit was not included for cross-connecting the service water discharge headers, (2) offsite power was assumed to be lost without the potential for recovery, and (3) HEPs for actions that were determined to be not feasible were always failed. The licensee explained that, although operator stress would be high, the impact of this factor on the HEP determination was not increased from the internal events value because the flooding hazard was caused by a hurricane-driven storm surge, which would occur with several hours of advance warning allowing sufficient time for feasible operator actions to be taken.

The licensee further stated that critical flood levels and propagation pathways for consideration in the XF PRA were determined from information gained during the walkdown and from plant-specific references. The licensee explained that design features were considered in the XF PRA and were credited for flood mitigation based on whether the feature would be expected to mitigate the flood under evaluation. The licensee cited plant procedures that identified the external flood protection features and the corresponding inspection process. The licensee stated that its procedures resulted in periodic inspections of all passive flood protection features

as well as manhole covers, active flood protection features (i.e., check valves, sump pumps, etc.), and temporary passive flood protection features such as barriers.

The NRC staff's review finds that the licensee performed an evaluation of the impact of an external flood on operator actions consistent with the relevant SRs in the 2009 ASME/ANS PRA Standard and the plant response conservatively did not credit cross-connecting the service water discharge headers, offsite power recovery, and actions that were determined not to be feasible. The staff's review further finds that the licensee's procedures lead to pre-staging multiple operators at different locations, as necessary, upon severe weather warning, and that procedures identify active as well as passive flood protection features for periodic inspections which supports the corresponding credit in the XF PRA. Therefore, the NRC staff finds that the licensee has dispositioned findings associated with evaluating the potential impact of XF on system recoveries (F&O XFPR-A11-1) and performing a systematic review of potential impacts of external flooding (F&O XFPR-A3-1) for this application. The NRC staff's walkdown of operator pathways and locations for dominant operator actions during the August 2018 onsite audit (Reference 10) provided confidence in the feasibility of those actions.

Finding XFPR-A7-1, related to SR XFPR-A7, called for the performance of an analysis of external hazard-caused dependencies. The resolution provided in the LAR stated that the XF analysis does not model dependencies and correlations of equipment failure other than the effects from inundation and that the analysis has equipment failure correlated due to submergence. The 2009 ASME/ANS PRA Standard states that spatial and environmental dependencies that can affect multiple SSCs or a combination of SSCs in the XF PRA model need to be considered. The resolution did not provide sufficient information to determine whether dependencies have been appropriately considered and included in the XF PRA model. In the November 2, 2018, supplement (Reference 2), the licensee stated that spatial and environmental dependencies in the XF PRA model were analyzed and provided details of such dependencies. Based on its analysis, the licensee concluded that such dependencies were either captured in the XF PRA or did not impact the XF PRA. The licensee explained that simultaneous consideration of HW and XF was included in the XF PRA and that the modeling of the failure of SSCs such as the SAMA DG and the EDG exhaust between the HW and XF PRAs was consistent. In the XF model, the HW drive a storm surge that leads to failure of these components by inundation whereas in the HW model, failure of the components is tied more directly to component fragility. The NRC staff's review of the LAR and the supplement finds that the licensee has dispositioned finding level F&O XFPR-A7-1 for this application because (1) the licensee performed an evaluation of the dependencies and correlations of equipment failure for the XF PRA and, as applicable, appropriately captured such dependencies in the XF PRA, and (2) simultaneous consideration of HW and XF was included in the development of the storm surge hazard used in the XF PRA. The NRC staff's conclusion on the development of the storm surge hazard used in the XF PRA is discussed in Section 3.5.1.4.3 of this SE.

Finding XFPR-C2-1, related to SR XFPR-C2, stated that the specific adaptations to the internal events PRA to produce the XF PRA was not documented. Since the documentation was unavailable at the time of the peer review, it appeared that the peer reviewers did not have information necessary to determine whether the adaptation of the internal events model was performed appropriately. In THE supplement dated November 2, 2018 (Reference 2), the licensee listed the changes that were made to the internal events PRA model to capture the impact of the 20-ft and 23-ft flood elevations. The licensee stated that the 20-ft flood failed the switchyard, the electric and diesel firewater pumps, and the circulating water pumps while the 23-ft flood failed the EDGs in addition to the SSCs failed at 20 ft. The NRC staff finds that the licensee has appropriately dispositioned the finding associated with adaptation of internal

events PRA (F&O XFPR-C2-1) for this application because the licensee used relevant parts of its internal events model [(i.e., LOOP event tree)] for the XF PRA and included modifications necessary to capture the occurrence and impact of XF. The NRC staff considered information related to finding level F&Os XFPR-A11-1, XFPR-A3-1, and XFPR-A7-1 (related to the analysis of dependencies and correlations caused by external flooding events) in reaching its conclusion on the disposition of finding level F&O XFPR-C2-1 for this application.

In summary, the NRC staff concludes that the licensee dispositioned all of the finding level F&Os for the XF PRA for this application based on the acceptability of the reported disposition for this application.

3.5.1.4.7 High Winds and External Flooding PRA Acceptability Conclusion

Pursuant to 10 CFR 50.69(c)(1)(i), the categorization process must consider results and insights from a plant-specific PRA. The use of HW and XF PRAs to support categorization is endorsed by RG 1.201, Revision 1. The PRAs must be acceptable to support the categorization process and must be subjected to a peer review process assessed against a standard that is endorsed by the NRC. Revision 2 of RG 1.200 provides guidance for determining the acceptability of the PRA by comparing the PRA to the relevant parts of the ASME/ANS 2009 Standard using a peer review process. The licensee has subjected its HW and XF PRAs to the peer review process and submitted the results of the peer review.

The NRC staff reviewed the peer review process and its results and findings, the licensee's resolution of peer review findings, the identification as well as disposition of key assumptions and sources of uncertainty. Based on its review, the NRC staff finds that the licensee's HW and XF PRA are acceptable to support the categorization of SSCs using the process endorsed by the NRC staff in RG 1.201 and key assumptions for the HW and XF PRAs are identified consistent with the guidance in RG 1.200, Revision 2, and NUREG-1855, as applicable, and addressed appropriately for this application. The NRC staff's review further finds that inclusion of the storm surge analysis in the XF PRA will be consistent with the guidance in RG 1.200, Revision 2, and the 2009 ASME/ANS PRA Standard endorsed therein subsequent to the completion of implementation item (i) in Attachment 1 of the enclosure to the April 8, 2019, supplement (Reference 4), as part of the proposed license condition. Implementation item (i) in Attachment 1 of the enclosure to the April 8, 2019, supplement necessitates a focused-scope peer review of the XF PRA model hazard development as well as the closure of any resulting findings per an NRC-approved process.

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 during categorization. These requirements are met, in part, by using importance measures and sensitivity studies, as described in the methodology in NEI 00-04, Section 5. Fussell-Vesely (F-V) and Risk Achievement Worth (RAW) importance measures are obtained for each component and each PRA modeled hazard (i.e., separately for the internal events PRA and for the fire PRA) and the values are compared to specified criteria in NEI 00-04. Components that have internal event importance measure values that exceed the criteria are assigned HSS and cannot be changed by the IDP. Components that have fire, HW, and XF event importance measures exceeding the criteria are assigned preliminary HSS. Integrated importance measures over all PRA modeled hazards are calculated per Section 5.6 of NEI 00-04, and components for which the integrated measures exceed the criteria are assigned preliminary HSS.

The guidance in NEI 00-04 specifies the 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 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, supplemented with EPRI TR-1016737, "Treatment of Parameter and Modeling Uncertainty for Probabilistic Risk Assessments," December 2008 (Reference 39), to identify sources of uncertainty in the internal events PRA. The NRC staff requested in PRA-RAI-04.a that the licensee describe the process utilized to determine the candidates for key sources of uncertainty and confirm that that this process meets the guidance in NUREG-1855, Revision 1. In its supplement dated November 2, 2018, the licensee provided a gap assessment of the process used for the LAR and NUREG-1855, Revision 1, and in conjunction with the request of PRA-RAI-04.b provided an updated list of key sources of uncertainty (Reference 2). The NRC staff requested further explanation and justification of the identification and treatment of key assumptions and sources of uncertainties in RAI 4.01/17.01.

In response to RAI 4.01/17.01, dated April 8, 2019 (Reference 4), the licensee stated that consistent with NUREG-1855, Revision 1, Stage E, it uses Table A.1 of EPRI TR-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, "Practical Guidance on the Use of Probabilistic Risk Assessment in Risk Informed Applications with a Focus on the Treatment of Uncertainty," December 2012 (Reference 40), as well as the PRA documentation for plant-specific assumptions and uncertainties. The response to RAI 4.01/17.01 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," October 2006 (Reference 41), 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.

Table 1 of the licensee's April 8, 2019, letter (Reference 4) lists and dispositions key sources of uncertainty identified by the licensee. For each of the identified key sources of uncertainty related to internal events and internal fires, the licensee has shown that the uncertainty or assumptions will not affect the 50.69 categorization results or that sensitivity studies will be performed as necessary in accordance with NEI 00-04 Section 5.

The NRC staff's review of the identification and disposition of the key assumptions and sources of uncertainty in the HW and XF PRAs is discussed in Section 3.5.1.4.5 of this SE.

Based on its review, the NRC staff finds that the licensee searched for, identified, and evaluated sources of uncertainty in its internal events (including internal flooding), fire, HW and XF PRAs consistent with the guidance in RG 1.200, Revision 2, NUREG-1855, and EPRI TR-1016737, as

applicable. Therefore, the NEI 00-04 guidance to identify additional "applicable sensitivity studies" is satisfied.

The categorization of SSCs using the licensee's HW and XF PRA models is expected to be based on importance measures and corresponding numerical criteria as described in Sections 5.1 and 5.3 of NEI 00-04. Further, Section 5.6 of NEI 00-04 discusses the integral assessment wherein the hazard-specific importance measures are weighted by the hazards contribution to the plant risk. The NRC staff requested the licensee to describe how the importance measures are determined from the HW and XF PRA models in the context of the "binning" approach. The NRC staff also requested the licensee to describe how the resulting importance measures are compared to the numerical criteria in NEI 00-04. In the supplement dated November 2, 2018 (Reference 2), the licensee stated that HW PRA used a "binning" approach to address the different probabilities of failure of some components due to the different initiating events. Therefore, there would be multiple basic events representing the HW-induced failure of those components that needed to be combined to develop importance measures. The licensee explained that based on the underlying layout of the HW PRA, the F-V value for any component from the HW PRA would be obtained by adding the F-V value for all basic events that represent the failure of that component, including both HW-induced failures and random failures. The licensee further explained that in order to calculate the RAW value for a component, all basic events that represented the different failure modes of that component would be failed (i.e., set to logical true) at the same time. The resulting risk metric (CDF or LERF) would then be divided by the baseline risk metric to obtain the RAW value for that component from the HW PRA. The F-V and RAW value for each component would then be directly compared to the numerical criteria in NEI 00-04.

The licensee stated that the XF PRA does not use the "binning" approach and considers a component to be failed when the component becomes submerged. As a result, importance measures for all components in the XF model would be determined from the random failure events for components in the model, just as they are for internal events, and the numerical criteria in NEI 00-04 would be applied the same way (i.e., sum of F-V, max of RAW, etc.) for the basic events that apply to each component.

Based on its review, the NRC staff finds that the licensee's approach for determining the HW and XF PRA-specific importance measures for basic events and calculating the corresponding integrated importance measures is consistent with the NRC-endorsed guidance of NEI 00-04. Therefore, the NRC staff concludes that the calculation and use of importance measures from the HW and XF PRA by the licensee is acceptable for this application.

The NRC staff requested information on the determination of the integrated importance measures for certain components where basic events, which represent different failure modes for a component, in the HW and XF PRA models may not align with basic events in other PRA models. The NRC staff specifically requested discussion of any mapping that would be performed between HW and XF PRA basis events and those in other PRA models and the treatment of implicitly modeled components in the HW and XF PRA models in the categorization process. In the supplement dated November 2, 2018 (Reference 2), the licensee stated that the integral assessment is performed on a component basis, not on a basic event basis. The licensee further stated that if a component is credited in one hazard model, but not in all (or any) of the other models, for those models which do not credit the component, the F-V value would be 0.0 and the RAW value would be 1.0. These values would then be used in the integral assessment formulas in Section 5.6 of NEI 00-04 with the CDF or LERF contribution from those models included in the denominator. The licensee explained that if a component was explicitly

modeled in one hazard model but was treated as being within the component boundary of another larger component (i.e., a 'subcomponent') in other models, the importance measures for the subcomponent would be calculated from the model where it was modeled explicitly and combined with the importance measures of the larger component to determine the importance of the larger component. The combination would be performed by adding the F-V values and using the maximum of the RAW values. The larger component would then be used in the integral assessment. The licensee further explained that in the categorization process, once a component becomes HSS for a function, all other components within that system which support that function also become HSS and, therefore, if the larger component became HSS, the subcomponent would also become HSS, such that a separate integral calculation for the subcomponent was not required.

In the supplement dated February 13, 2019 (Reference 3), the licensee stated that since plant components could not initiate external events such as tornadoes, HW, or XF, SSCs were not implicitly modeled as part of initiating events for the HW and XF PRAs. The licensee explained that some SSCs that were not explicitly modeled are considered subcomponents of other SSCs that were explicitly modeled. In such cases, importance measures were not developed for the subcomponents, but the subcomponents were given the same risk categorization as the highest risk component that was required to support the function of the explicitly modeled component. Such a determination was made after the integrated assessment was performed for the explicitly modeled components. The licensee further explained that the categorization process in NEI 00-04 allowed the subcomponent to be downgraded to a lower risk category if it was demonstrated that the subcomponent was not required to support the modeled function. On the topic of implicitly modeled SSCs that were not a subcomponent of an explicitly modeled component, but must perform their function in order for that explicitly modeled component to perform its function (for example, a floor drain that must remove water during an XF event to prevent failure of another component within the room), the licensee clarified that the implicitly modeled component was given the same risk significance as the explicitly modeled component that it supported. For SSCs that are implicitly modeled within an event representing a human action (i.e., the SSC is necessary for the human action to be successful), if the failure of the SSC by itself would prevent the successful completion of a risk significant human action, the SSC would become candidate HSS, regardless of the PRA from which the human action arises. If the failure of the SSC by itself would not prevent successful completion of the human action, it is considered LSS. The NRC staff's review of the discussion in the supplement finds that the licensee's process for capturing and treating implicitly modeled components in the HW and XF PRAs to be acceptable for this application because (1) it appropriately follows the guidance in Sections 5.4, 7.1, and 10.2 of NEI 00-04 for identifying and considering implicitly modeled components in the categorization process, and (2) SSCs that are not subcomponents of an explicitly modeled component but must perform their function in order for that explicitly modeled component to perform its function would have the same risk significance, and therefore, categorization, as the explicitly modeled component that it supports ensuring that the function of the implicitly modeled components is captured.

In response to the NRC staff's request for a description of how sufficient data points for the XF hazard were determined to capture the plant response at different flooding elevations, the licensee stated that a cliff-edge effect, which is caused by the failure of DGs, in the plant response occurs at an elevation of 23 ft and that the majority of plant risk in response to XF events occurs at an elevation at and above 20 ft but below 23 ft. As noted by the licensee's response, the plant response will be different at 23 ft as compared to 20 ft. Further, the failure of the DGs at 23 ft can affect this application (e.g., SSCs, such as the SAMA DGs, becoming HSS) which can be missed if that flood elevation is not quantified as part of the base XF PRA.

In its letter dated February 13, 2019 (Reference 3), the licensee stated that the modeling of the 23-ft flood assumed an unrecoverable LOOP, the loss of the circulating water pumps and fire pumps (both diesel and electric pumps), the failure of the EDGs, and failure of the operator action to start the SAMA diesels that charge the station batteries. The licensee explained that the operator action to start the SAMA diesels was failed due to the inability to access the area necessary to perform the action. The licensee stated that due to the unrecoverable loss of offsite power and lack of onsite emergency power, suppression pool cooling and wetwell venting would be unavailable preventing a safe and stable condition to be achieved. Due to the equipment assumed to be failed for a 23-ft flood and the resulting plant response, the licensee stated that there were no unique risk insights to be gained by including the 23-ft flood evaluation in the categorization process. The NRC staff reviewed the licensee's response in conjunction with information about the XF PRA in the supplements. The NRC staff's review notes that (1) the licensee used relevant part of the internal events PRA (i.e., LOOP event tree) to build the XF PRA and did not include additional equipment besides that already modeled in the internal events PRA, and (2) although water accumulation until an elevation of 5 ft in the DG building basement is necessary for failure of all DGs, the XF PRA modeling at the 23-ft elevation assumes failure of all EDGs instantaneously at the onset of the flood. Based on the review of the information in the November 2, 2018 (Reference 2), and the February 13, 2019 (Reference 3), supplements, the NRC staff finds that the 23-ft flood is not expected to add any unique insights or unique HSS SSCs compared to the internal events PRA as well as the XF PRA for the 20-ft flood.

3.5.3 Non-PRA Methods

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

As described in the LAR, as supplemented, 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 42) to assess shutdown risk.

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

<u>Seismic Risk</u>

To assess seismic risk for the 10 CFR 50.69 categorization process, the licensee will use the SMA method. The SMA is a screening method that does not quantify CDF. The licensee used the SMA method during its IPEEE in response to Generic Letter 88-20 (Reference 43). The SMA method includes the development of the seismic SSEL, which contains the components that would be needed during and after a seismic event. The SSEL identifies one preferred and one alternate path capable of achieving and maintaining safe shutdown conditions for at least 72 hours following an earthquake. The licensee stated in Section 3.2.3 of the LAR that it will

follow the NEI 00-04 approach using the SSEL to identify credited equipment as HSS regardless of their 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, the licensee's 10 CFR 50.69 categorization process considers all components in the SSEL as HSS based on seismic risk.

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

Other External Hazards

The licensee stated that external hazards were initially evaluated by the licensee during the IPEEE. This hazard category includes external hazards such as 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.

In Section 3.2.4 of the LAR, the licensee stated that an evaluation will be conducted to determine if a scenario was screened due to an SSC in accordance with the flow chart in Figure 5-6 in Section 5.4 of NEI 00-04 and that, if a hazard was screened from applicability, the related SSC would be considered insignificant. In response to PRA-RAI-07 (Reference 2), the licensee clarified that 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 HSS.

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

Shutdown Risk

Paragraph 50.69(c)(1)(ii) of 10 CFR requires the licensee to determine SSC functional importance using an integrated, systematic process for addressing initiating events (internal and external), SSCs, and plant operating modes, including those not modeled in the plant-specific PRA. Consistent with the NEI 00-04 guidance, the licensee proposes to use the shutdown safety assessment process based on NUMARC 91-06. The guidance in NUMARC 91-06 provides considerations for maintaining DID for the five key safety functions during shutdown, namely, decay heat removal capability, inventory control, power availability, reactivity control, and containment - primary/secondary. The guidance in NUMARC 91-06 specifies that a DID approach should be used with respect to each defined shutdown key safety function. This is accomplished by designating a running and an alternative system/train to accomplish the given key safety function.

The licensee's process is consistent with the guidance in NEI 00-04, Section 5. The licensee indicated that components are categorized with respect to shutdown risk using a non-PRA shutdown assessment as follows:

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

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

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

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

3.5.4 Component Safety Significance Assessment for Passive Components

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

In the LAR, the licensee proposed using a categorization method for passive components not cited in NEI 00-04 for passive component categorization, but approved by the NRC on April 22, 2009, for Arkansas Nuclear One, Unit 2 (ANO-2) (Reference 44). The ANO-2 methodology is an RI safety classification and treatment program for repair/replacement activities for Class 2 and Class 3 pressure-retaining items and their associated supports (exclusive of Class CC and MC items), using a modification of the ASME Code Case N-660, "Risk-Informed Safety Classification for Use in Risk-Informed Repair/Replacement Activities, Section XI, Division 1," July 2002 (Reference 45). The ANO-2 methodology relies on the conditional core damage and large early release probabilities associated with pipe ruptures. Safety significance is generally measured by the frequency and the consequence of, in this case, pipe ruptures. Treatment requirements (including repair/replacement) only affect the frequency of passive component failure. Categorizing solely based on consequences, which measures the safety significance of the pipe given that it ruptures, is conservative compared to including the rupture frequency in the categorization. The categorization will not be affected by changes in frequency arising from changes to the treatment.

In Section 3.1.2 of the LAR, the licensee stated that it will apply the ANO-2 methodology to ASME Class 1 SSCs since the consequence evaluation and deterministic considerations are independent of the ASME classification when determining the SSCs safety significance. However, the ANO-2 passive categorization methodology excluded all Class 1 pressure boundary components and limits the scope of this process to Class 2 and Class 3 SSCs. In its

November 2, 2018, response to PRA-RAI-03 (Reference 2), the licensee confirmed that all ASME Code Class 1 SSCs with a pressure-retaining function, as well as supports, will be assigned HSS.

Because all Class 1 SSCs and supports will be considered HSS, and only Class 2 and Class 3 SSCs will be categorized using the ANO-2 passive categorization methodology consistent with previous NRC staff approval, the NRC staff finds the licensee's proposed approach for passive categorization acceptable for the 10 CFR 50.69 categorization process.

3.5.5 Summary

The NRC staff reviewed the PRA and the non-PRA methods used by the licensee in its 10 CFR 50.69 categorization process to assess the safety significance of active and passive components and finds these methods acceptable and consistent with RG 1.201 and the NRC endorsed guidance in NEI 00-04. 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 risk and internal flooding
- HW PRA to assess HW risk
- XF PRA to assess XF risk
- Fire PRA to assess fire risk
- SMA to assess seismic risk
- Screening using IPEEE to assess risk from other external hazards
- Shutdown safety assessment process to assess shutdown risk
- ANO-2 (see Reference 44) 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 NRC staff's RAIs, the NRC staff identified certain specific actions necessary to support their conclusion that the proposed program meets the requirements in 10 CFR 50.69 and the guidance in RG 1.201 and NEI 00-04. The licensee proposed the addition of a license condition for the implementation of 10 CFR 50.69. The license condition identifies three implementation items that shall be completed prior to the implementation of the 10 CFR 50.69 categorization process:

i. Duke Energy will complete a focused scope peer review of the BSEP External Flood PRA model hazard development prior to implementation of 10 CFR 50.69. Any findings from the focused scope peer review will be resolved and closed per an NRC accepted process prior to implementing 10 CFR 50.69.

ii. Duke Energy will update the applicable PRA models with FLEX DG failure rates as described in Attachment 3 of Duke Energy letter dated April 8, 2019 prior to implementing 10 CFR 50.69.

iii. The operator actions and associated equipment failures modeling containment venting will be added to the BSEP LERF model as described in response to RAI 4.01 / 17.01 in Duke Energy letter dated April 8, 2019 prior to implementing 10 CFR 50.69.

The NRC staff notes that either a focused-scope peer review or the IAT F&O closure process accepted by the NRC via letter dated May 3, 2017 (Reference 25) can be used by the licensee for completing implementation item (i), as part of the proposed license condition, in Attachment 1 of Duke Energy letter dated April 8, 2019 (Reference 4), which is reproduced above.

3.6 Assessment of Defense-in-Depth and Safety Margins

Paragraph 50.69(c)(1)(iii) of 10 CFR requires that the process used for categorizing SSCs must maintain DID. NEI 00-04, 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 of NEI 00-04 to be categorized as HSS.

Figure 6-1 in NEI 00-04 provides guidance to assess design-basis DID based on the frequency of the design-basis initiating event and the number of redundant and diverse trains nominally available to mitigate the initiating event. For each initiating event frequency, components are assigned as HSS if fewer than the indicated number of mitigating trains are nominally available. Section 6 of NEI 00-04 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. The DID for beyond design-basis initiating events is addressed by the PRA categorization process.

RG 1.201 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 containment penetrations and valves may be exempted from the Type Band Type C leakage testing requirements in both Options A and B of Appendix J to 10 CFR Part 50, but the 10 CFR 50.69(b)(1)(x) criteria are not used to determine the proper RISC category for containment isolation valves or penetrations.

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

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

3.7 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. In addition, any component identified as HSS from either the integrated risk component safety significance assessment or the DID assessment cannot be changed by the IDP.

The NRC staff finds that the licensee's preliminary categorization process is consistent with the guidance in NEI 00-04, as endorsed in RG 1.201 and, therefore, acceptable.

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

Paragraph 50.69(c)(1)(iv) of 10 CFR requires that any potential increases in CDF and LERF resulting from changes to treatment are small. The guidance in Section 8 of NEI 00-04, as endorsed by RG 1.201, 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). Section 3.1.1 and Section 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).

Section 8 of NEI 00-04 does not include explicit discussion of risk sensitivity studies for external hazard PRAs. The categorization of SSCs using the external hazard PRAs is dominated by failure modes that are dependent on the corresponding modeling inputs such as the 'dominant failure modes' and 'fragility curves.' These modeling inputs are derived using several parameters that can be impacted by alternative treatments. Therefore, the NRC staff requested the licensee to describe and justify how the required risk sensitivity study outlined in Section 8 of NEI 00-04 will be performed for categorization using the licensee's HW and XF PRA models to meet the requirements of 10 CFR 50.69(c)(1)(iv) and 10 CFR 50.69(b)(2)(iv). The NRC staff further requested the licensee to describe how it will be determined that the modeling inputs in the licensee's HW and XF PRA models and those used for the risk sensitivity study continue to remain valid to ensure compliance with the requirements of 10 CFR 50.69(e).

In the supplement dated November 2, 2018 (Reference 2), the licensee explained that since the XF PRA does not use a "binning" approach and considers a component failed when the component becomes submerged, the risk sensitivity study outlined in Section 8 of NEI 00-04 for the XF model would be performed in the same manner as for internal events (i.e., by increasing the random failure probability of all LSS components by a factor of 3). The licensee proposed not increasing the probability of the HW-induced failure events associated with the LSS components for the risk sensitivity study outlined in Section 8 of NEI 00-04 based on the programs and processes which, the licensee stated, provided reasonable confidence that the wind capacities of LSS component would not be impacted by alternative treatments. The licensee explained that, as stated in 10 CFR 50.69, it must be ensured, with reasonable confidence, that RISC-3 SSCs (i.e., SSCs that are safety-related and categorized as LSS) remain capable of performing their safety-related functions under design-basis conditions, including environmental conditions and effects throughout their service life. The licensee further explained that periodic inspection and testing activities must be conducted to determine that RISC-3 SSCs remained capable of performing their safety-related functions under design-basis conditions. The licensee stated that any identified degradation would be corrected by its processes. The licensee further stated that its configuration management program would maintain the configuration of SSCs in the plant and that whenever a change was made to an SSC, an appropriate design change process would be utilized to ensure that design requirements remain unchanged as required by the 10 CFR 50.69 rule. The NRC staff's review finds that (1) the licensee's current programs provide reasonable assurance that the high-wind capacities of LSS components would not be significantly impacted and (2) the monitoring and corrective action programs for SSCs ensures that potential degradation of the wind capacity would be detected and addressed before significantly impacting the HW risk. Because of the licensee's programs and processes, the exclusion of the high-wind capacity of LSS components in the risk sensitivity study is acceptable for this application. Because of the approach used in the XF PRA to model the SSCs impacted by XF events, there is no XF capacity that needs to be addressed and the risk sensitivity study is appropriately performed for the XF PRA.

This sensitivity study, together with the periodic review process reviewed in Section 3.10 of this SE, assures that the potential cumulative risk increase from the categorization is maintained acceptably low. 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 Decision-making Panel Review and Approval (NEI 00-04, Sections 9 and 10)

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

The guidance in NEI 00-04, endorsed in RG 1.201, 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, as endorsed by RG 1.201. Therefore, all aspects of the integrated, systematic process used to characterize SSCs will reasonably reflect current plant configuration and operating practices, and applicable plant and industry operational experience as required by 10 CFR 50.69(c)(1)(ii).

The IDP may change the categorization of a component from LSS to HSS based on its assessment and decision-making. As outlined in NEI 00-04, Section 10.2, the IDP may re-categorize components supporting an HSS function from HSS to LSS only if a credible failure of the component would not preclude the fulfillment of the HSS function and the component was not categorized as HSS based on the six criteria above (i.e., internal events PRA, integrated PRA component risk, SMA, shutdown, passive categorization, and DID).

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, as endorsed in RG 1.201, provides guidance on program documentation and change control, and Section 12 provides guidance on periodic review. These sections are described in NEI 00-04 with respect to satisfying 10 CFR 50.69(e) and 10 CFR 50.69(f), respectively. Maintaining change control and periodic review will also maintain confidence that all aspects of the program reflect current plant operation.

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

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

In its November 2, 2018, response to PRA-RAI-06 (Reference 2), the licensee explained that the review would be conducted at least once every other fuel cycle and be conducted by a senior engineer and a PRA engineer. The NRC staff finds that this description is consistent with the requirements for feedback and process adjustment required by 10 CFR 50.69(e), and is, therefore, acceptable.

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

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

In addition, LAR Attachment 1 (List of Categorization Prerequisites) states that the licensee will establish procedures 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 review to ensure continued categorization validity and acceptable performance for SSCs that have been categorized, and (8) documentation requirements identified in LAR Section 3.1.1. Procedures are formal plant documents and changes will be tracked providing change control and records of the changes.

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

In the supplement dated November 2, 2018 (Reference 2), the licensee stated that its PRA maintenance process included consideration of changes in external events hazards when completing a model update. The licensee explained that the process would consider changes to the plant-specific hazard inputs (e.g., topographical changes that might impact flood runup) as well as changes to the generic hazard frequencies and hazard strength data. The licensee also stated that any new industry guidance on hazard development methodology would be considered as specifically required by the licensee's PRA maintenance procedures. The licensee further explained that its procedures specifically address physical plant changes and that following a PRA model update, the procedures required that PRA applications, including 10 CFR 50.69, be reviewed to assess the impact of the model change on the application. The licensee stated that its procedures and the impact assessment following model changes would address any plant changes as well as other modeling inputs.

Based on its review of the submittal and supplements, 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 NRC staff finds the process in NEI 00-04 and the LAR will meet the requirements of 10 CFR 50.69(e) and 10 CFR 50.69(f), respectively. Therefore, the process used to characterize SSC importance will reasonably reflect the current plant configuration and operating practices, and applicable plant and industry operational experience required in 10 CFR 50.69(c)(1)(ii).

3.11 Technical Conclusion

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

(1) considers results and insights from plant-specific internal events (including internal flooding), fire, HW, and XF PRAs, which are of sufficient quality and level of detail to support the categorization process and that either have been subjected to a peer review process against RG 1.200 Revision 2 or will be subjected to such a process prior to implementation of the 10 CFR 50.69 program, 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; and

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

4.0 10 CFR 50.69 IMPLEMENTATION LICENSE CONDITION

Section 50.69(b)(2) of 10 CFR requires the licensee to submit an application that describes the categorization process. Section 50.69(b)(3) of 10 CFR states that the Commission will approve the license application if it determines that the categorization process satisfies the requirements of 10 CFR 50.69(c). As described in this SE, the NRC staff has concluded that the 10 CFR 50.69 categorization process described in the licensee's application, 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 NRC staff's conclusion that the proposed program meets the requirements in 10 CFR 50.69 and the guidance in RG 1.201 and NEI 00-04.

The NRC staff's finding on the acceptability of the PRA evaluation in the licensee's proposed 10 CFR 50.69 process is conditioned on the completion of three implementation items. All of these items are identified in Attachment 1 of the licensee's letter dated April 8, 2019 (Reference 4). The NRC staff notes that the licensee described some additional minor changes to the PRA and PRA methods. The NRC staff determined that these minor changes would not impact the 10 CFR 50.69 categorization process and were similar to occasional future changes to the PRA that occur over time. Therefore, the NRC staff determined that these additional minor changes do not need to be resolved prior to implementation of the 10 CFR 50.69 categorization process and, therefore, can be addressed and resolved using the licensee's periodic review process.

In Attachment 2 to its April 8, 2019 letter (Reference 4), the licensee proposed the following condition to its licenses:

> 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 1 [Unit 2] License Amendment No. [XXX] dated [DATE].

Duke Energy will complete the implementation items list in Attachment 1 of Duke letter to NRC dated April 8, 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).

Based on its evaluation in this SE, the NRC staff finds that the proposed license condition and its referenced implementation items are acceptable because they adequately implement 10 CFR 50.69 using models, methods, and approaches consistent with the applicable guidance that has previously been endorsed as acceptable by the NRC. For each implementation item, the licensee and the NRC staff have reached a satisfactory resolution involving the level of detail and main attributes that each remaining item will incorporate into the program upon its completion. Completion of these items does not change or impact the bases for the safety conclusions made by the NRC staff in 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 North Carolina State official on May 28, 2019, of the proposed issuance of the amendments. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, published in the *Federal Register* on May 22, 2018

(83 FR 23731), and there has been no public comment on such finding. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

7.0 CONCLUSION

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

8.0 REFERENCES

1. Gideon, W. R., Duke Energy Progress, LLC, letter to U.S. Nuclear Regulatory Commission, "Application to Adopt 10 CFR 50.69, 'Risk-Informed Categorization and Treatment of Structures, Systems, and Components (SSCs) for Nuclear Power Reactors,'" dated January 10, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18010A344).

2. Donahue, J., Duke Energy Progress, LLC, letter to U. S. Nuclear Regulatory Commission, "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 2, 2018 (ADAMS Accession No. ML18306A523).

3. Snider, S., Duke Energy Progress, LLC, letter to U. S. Nuclear Regulatory Commission, "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 February 13, 2019 (ADAMS Accession No. ML19044A366).

4. Snider, S., Duke Energy Progress, LLC, letter to U. S. Nuclear Regulatory Commission, "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 April 8, 2019 (ADAMS Accession No. ML19099A035).

5. Galvin, D., U.S. Nuclear Regulatory Commission, e-mail to Zaremba, A., Duke Energy, "RE: Brunswick - Adoption of 10 CFR 50.69 - HWPRA and XFPRA Audit Plan and Setup of Online Reference Portal (EPID: L-2018-LLA-0008)," dated May 9, 2018 (ADAMS Accession No. ML18130A021).

6. Barillas, M., U.S. Nuclear Regulatory Commission, letter to Capps, S., Duke Energy Corporation, "Brunswick Steam Electric Plant, Units 1 and 2, and Shearon Harris Nuclear Power Plant, Unit 1 - Audit Plan RE: Application to Adopt 10 CFR 50.69, 'Risk-Informed Categorization and Treatment of Structures, Systems, and Components (SSCs) for Nuclear Power Reactors,'" dated July 2, 2018 (ADAMS Accession No. ML18180A418).

7. Barillas, M., U.S. Nuclear Regulatory Commission, letter to Capps, S., Duke Energy Corporation, "Brunswick Steam Electric Plant, Units 1 and 2, and Shearon Harris Nuclear Power Plant, Unit 1 - Audit Plan Questions RE: Application to Adopt 10 CFR 50.69, 'Risk-Informed Categorization and Treatment of Structures, Systems, and Components (SSCs) for Nuclear Power Reactors,'" dated July 11, 2018 (ADAMS Accession No. ML18191A997).

8. Galvin, D., U.S. Nuclear Regulatory Commission, e-mail to Zaremba, A., Duke Energy, "Brunswick - Adoption of 10 CFR 50.69 - Supplement to the HWPRA and XFPRA Audit Plan for BSEP Onsite Audit (EPID: L 2018-LLA-0008)," dated August 7, 2018 (ADAMS Accession No. ML18220A790).

9. Barillas, M., U.S. Nuclear Regulatory Commission, letter to Capps, S., Duke Energy Corporation, "Brunswick Steam Electric Plant, Units 1 and 2, and Shearon Harris Nuclear Power Plant, Unit 1 - Regulatory Audit Summary Regarding License Amendment Requests to Adopt 10 CFR 50.69, 'Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Plants' (EPID L-2018-LLA-0008 and EPID L-2018-LLA-0034)," dated November 14, 2018 (ADAMS Accession No. ML18282A224).

10. Galvin, D., U.S. Nuclear Regulatory Commission, letter to Gideon, W.R., Duke Energy Progress, LLC, "Brunswick Steam Electric Plant, Units 1 and 2 - Regulatory Audit Summary Re: License Amendment Request to Adopt 10 CFR 50.69, 'Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Plants' (EPID L-2018-LLA-0008)," dated July 1, 2019 (ADAMS Accession No. ML19149A210).

11. Galvin, D., U.S. Nuclear Regulatory Commission, e-mail to Zaremba, A., Duke Energy, "Brunswick RAIs - LAR to Allow Implementation of the Provisions 10 CFR 50.69 (EPID L-2018-LLA-0008)," dated October 9, 2018 (ADAMS Accession No. ML18282A149).

12. Galvin, D., U.S. Nuclear Regulatory Commission, e-mail to Zaremba, A., Duke Energy, "Brunswick 2nd Round RAIs - LAR to Allow Implementation of the Provisions 10 CFR 50.69 (EPID L-2018-LLA-0008)," dated January 14, 2019 (ADAMS Accession No. ML19015A030).

13. Galvin, D., U.S. Nuclear Regulatory Commission, e-mail to Zaremba, A., Duke Energy, "Brunswick 3rd Round RAI - LAR to Allow Implementation of the Provisions 10 CFR 50.69 (EPID L-2018-LLA-0008)," dated March 7, 2019 (ADAMS Accession No. ML19067A271).

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

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

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

17. American Society of Mechanical Engineers/American Nuclear Society, ASME/ANS RA-Sa-2009, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," February 2009 (ADAMS Accession No. ML092870592).

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

19. Hon, A., U.S. Nuclear Regulatory Commission, letter to Gideon, W. R., Duke Energy Progress, LLC, "Brunswick Steam Electric Plant, Units 1 and 2 - Issuance of Amendments Regarding Request to Relocate Specific Surveillance Frequencies to Licensee Controlled Program (CAC No. MF7206 and MF7207)," dated May 24, 2017 (ADAMS Accession No. ML17096A129).

20. Hon, A., U.S. Nuclear Regulatory Commission, letter to Gideon, W. R., Duke Energy Progress, Inc., "Brunswick Steam Electric Plant, Units 1 and 2 - Issuance of Amendment Regarding Transition to a Risk-Informed, Performance-Based Fire Protection Program in Accordance With 10 CFR 50.48(c) (TAC Nos. ME9623 and ME9624)," dated January 28, 2015 (ADAMS Accession No. ML14310A808).

21. Andersen, V. K., Nuclear Energy Institute, letter to Rosenberg, S., U.S. Nuclear Regulatory Commission, "Final Revision of Appendix X to NEI 05-04/07-12/12-16, Close Out of Facts and Observations (F&Os)," dated February 21, 2017 (ADAMS Package Accession No. ML17086A431).

22. Nuclear Energy Institute, NEI 05-04, "Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard," Revision 2, November 2008 (ADAMS Accession No. ML083430462).

23. Nuclear Energy Institute, NEI 07-12, "Fire PRA Peer Review Process Guidelines," Revision 1, June 2010 (ADAMS Package Accession No. ML102230049).

24. Nuclear Energy Institute, NEI 12-13, "External Hazards PRA Peer Review Process Guidelines," August 2012 (ADAMS Package Accession No. ML122400044).

25. Giitter, J., and Ross-Lee, M. J., U.S. Nuclear Regulatory Commission, letter to Krueger, G., Nuclear Energy Institute, "U.S. Nuclear Regulatory Commission Acceptance on Nuclear Energy Institute Appendix X to Guidance 05-04, 07-12, and 12-13, Close-Out of Facts and Observations (F&Os)," dated May 3, 2017 (ADAMS Accession No. ML17079A427).

26. Electric Power Research Institute, EPRI TR-1021081, "Establishing Minimum Acceptable Values for Probabilities of Human Failure Events," October 2010.

27. U.S. Nuclear Regulatory Commission, NUREG-1792, "Good Practices for Implementing Human Reliability Analysis (HRA)," April 2005 (ADAMS Accession No. ML051160213).

28. Reisi-Fard, M., U.S. Nuclear Regulatory Commission, memorandum to Giitter, J., U.S. Nuclear Regulatory Commission, "Assessment of the Nuclear Energy Institute 16-06, 'Crediting Mitigating Strategies in Risk-Informed Decision Making,'" dated May 30, 2017 (ADAMS Accession No. ML17031A269).

29. U.S. Nuclear Regulatory Commission, NUREG/CR-6928, "Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants," February 2007 (ADAMS Accession No. ML070650650).

30. U.S. Nuclear Regulatory Commission, NUREG/CR-4772, "Accident Sequence Evaluation Program Human Reliability Analysis Procedure," February 1987.

31. U.S. Nuclear Regulatory Commission, NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities, Volume 1: Summary & Overview," September 2005 (ADAMS Accession No. ML052580075).

32. U.S. Nuclear Regulatory Commission, NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities, Volume 2: Detailed Methodology," September 2005 (ADAMS Accession No. ML052580118).

33. Nuclear Energy Institute, NEI 00-02, "Probabilistic Risk Assessment Peer Review Process Guidance," Revision 1, May 2006 (ADAMS Accession No. ML061510619).

34. Franovich, M., U.S. Nuclear Regulatory Commission, letter to Kruger, G., Nuclear Energy Institute, "U.S. Nuclear Regulatory Commission Acceptance of Nuclear Energy Institute (NEI) Guidance NEI 12-13, 'External Hazards PRA Peer Review Process Guidelines, (August 2012),'" dated March 7, 2018 (ADAMS Accession No. ML18025C025).

35. Uribe, J., U.S. Nuclear Regulatory Commission, letter to Gideon, W. R., Duke Energy Progress, LLC, "Brunswick Steam Electric Plant, Units 1 and 2 - Interim Staff Response to Reevaluated Flood Hazards Submitted in Response to 10 CFR 50.54(f) Information Request - Flood Causing Mechanism Reevaluation (CAC Nos. MF6104 and MF6105)," dated March 16, 2017 (ADAMS Accession No. ML17072A364).

36. Carolina Power & Light Company, "Brunswick Nuclear Plant Individual Plant Examination for External Events Submittal Final Report," June 1995 (ADAMS Legacy Accession No. 9507060396).

37. Rubenstein, L. S., U.S. Nuclear Regulatory Commission memorandum to Miraglia, F., U.S. Nuclear Regulatory Commission, "Safety Evaluation Report - Electric Power Research Institute (EPRI) Topical Reports Concerning Tornado Missile Probabilistic Risk Assessment (PRA) Methodology," dated October 26, 1983 (ADAMS Accession No. ML080870291).

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

39. Electric Power Research Institute, EPRI TR-1016737, "Treatment of Parameter and Modeling Uncertainty for Probabilistic Risk Assessments," December 2008.

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

41. Electric Power Research Institute, EPRI TR-1013491, "Guideline for the Treatment of Uncertainty in Risk-Informed Applications," October 2006.

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

43. U.S. Nuclear Regulatory Commission, Generic Letter 88-20, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10 CFR 50.54(f)," Supplement 4, June 1991 (https://www.nrc.gov/reading-rm/doc-collections/gen-comm/gen-letters/1988/gl88020s4.html).

44. Markley, M. T., U.S. Nuclear Regulatory Commission, letter to Entergy Operations, Inc., "Arkansas Nuclear One, Unit 2 - Approval of Request for Alternative ANO2-R&R-004, Revision 1, Request to Use Risk-Informed Safety Classification and Treatment for Repair/Replacement Activities in Class 2 and 3 Moderate and High Energy Systems (TAC No. MD5250)," dated April 22, 2009 (ADAMS Accession No. ML090930246).

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

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