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

November 26, 2018

Mr. Mano Nazar
President and Chief Nuclear Officer
Nuclear Division
NextEra Energy Point Beach, LLC
Mail Stop: NT3/JW
15430 Endeavor Drive
Jupiter, FL 33478

SUBJECT: POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS TO ADOPT TITLE 10 OF THE CODE OF FEDERAL REGULATIONS SECTION 50.69, "RISK-INFORMED CATEGORIZATION AND TREATMENT OF STRUCTURES, SYSTEMS AND COMPONENTS FOR NUCLEAR POWER REACTORS" (CAC NOS. MG0196 AND MG0197) (EPID L-2017-LLA-0284)

Dear Mr. Nazar:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment Nos. 262 and 265 to Renewed Facility Operating License Nos. DPR-24 and DPR-27 for the Point Beach Nuclear Plant (PBNP), Units 1 and 2. The amendments consist of changes to the Renewed Facility Operating Licenses for PBNP in response to your application dated August 31, 2017, as supplemented by letters dated October 26, 2017, August 10, 2018, and September 28, 2018.

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

A copy of the related safety evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

Mahesh L. Chawla, Project Manager
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-266 and 50-301

Enclosures:
1. Amendment No. 262 to DPR-24
2. Amendment No. 265 to DPR-27
3. Safety Evaluation

cc: ListServ

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

NEXTERA ENERGY POINT BEACH, LLC

DOCKET NO. 50-266

POINT BEACH NUCLEAR PLANT, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 262
Renewed License No. DPR-24

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

   A. The application for amendment by NextEra Energy Point Beach, LLC (the licensee), dated August 31, 2017, as supplemented by letters dated October 26, 2017, August 10, 2018, and September 28, 2018, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Renewed Facility Operating License No. DPR-24 as indicated in the attachment to this license amendment.

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

FOR THE NUCLEAR REGULATORY COMMISSION

David J. Wrona, Chief
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility
Operating License

Date of issuance: November 26, 2018

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

NEXTERA ENERGY POINT BEACH, LLC

DOCKET NO. 50-301

POINT BEACH NUCLEAR PLANT, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 265
Renewed License No. DPR-27

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

   A. The application for amendment by NextEra Energy Point Beach, LLC (the licensee), dated August 31, 2017, as supplemented by letters dated October 26, 2017, August 10, 2018, and September 28, 2018, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Renewed Facility Operating License No. DPR-27 as indicated in the attachment to this license amendment.

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

FOR THE NUCLEAR REGULATORY COMMISSION

David J. Wrona, Chief
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility
Operating License

Date of issuance: November 26, 2018

ATTACHMENT TO LICENSE AMENDMENT NO. 262

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-24

AND LICENSE AMENDMENT NO. 265

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-27

FOR

POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-266 AND 50-301

Replace the following pages of Renewed Facility Operating License Nos. DPR-24 and DPR-27, with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Renewed Facility Operating License

REMOVE	INSERT
Page 4	Page 4
Page 5	Page 5
Page 6	Page 6
Page 7	Page 7
Page 8	Page 8
-	Page 9

D. Physical Protection

NextEra Energy Point Beach shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans, including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contain Safeguards Information protected under 10 CFR 73.21, is entitled: "Point Beach Nuclear Plant Physical Security Plan, (Revision 4)," submitted by letter dated May 10, 2006. NextEra Energy Point Beach, LLC shall fully implement and maintain in effect all provisions of the Commission-approved Point Beach Nuclear Plant Cyber Security Plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The NextEra Energy Point Beach CSP was approved by License Amendment No. 243 as supplemented by a change approved by License Amendment No. 247 and License Amendment No. 252.

E. Adoption of 10 CFR 50.69, "Risk-Informed categorization and treatment of structures, systems, and components for nuclear power plants"

1. NextEra Energy Point Beach is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 structures, systems, and components (SSCs) using: Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events, including internal flooding, and internal fire; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (AN0-2) 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 License Amendment No. 262 dated November 26, 2018.

2. Prior to implementation of the provisions of 10 CFR 50.69, NextEra Energy Point Beach shall complete the items below:

   a. Item A in Attachment 1, List of Categorization Prerequisites, to NextEra Energy Point Beach letter NRC 2017-0043, "License Amendment Request 287, Application to Adopt 10 CFR 50.69, 'Risk-Informed Categorization and Treatment of Structures, Systems, and Components (SSCs) for Nuclear Power Plants,' " dated August 31, 2017; and

   b. Attachment 1, Point Beach 10 CFR 50.69 PRA Implementation Items, in NextEra Energy Point Beach letter NRC-2018-0044, "Supplement to Response to Request for Additional Information Regarding License Amendment Request 287, Application to Adopt 10 CFR 50.69, 'Risk informed Categorization and Treatment of Structures, System, and Components (SSCs) for Nuclear Power Plants,' " dated September 28, 2018.

3. 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).

F. NextEra Energy Point Beach Unit 1 shall implement and maintain in effect all provisions of the approved fire protection program that comply with 10 CFR 50.48(a) and 10 CFR 50.48(c), as specified in the license amendment request dated June 26, 2013, and supplements dated September 16, 2013, July 29, 2014, August 28, 2014, September 25, 2014, November 14, 2014, December 19, 2014, January 16, 2015, May 12, 2015, August 26, 2015, February 22, 2016, April 07, 2016, and May 3, 2016, and as approved in the safety evaluation report dated September 8, 2016. Except where NRC approval for changes or deviations is required by 10 CFR 50.48(c), and provided no other regulation, technical specification, license condition or requirement would require prior NRC approval, the licensee may make changes to the fire protection program without prior approval of the Commission if those changes satisfy the provisions set forth in 10 CFR 50.48(a) and 10 CFR 50.48(c), the change does not require a change to a technical specification or license condition, and the criteria listed below are satisfied.

1. Risk-Informed Changes that May Be Made Without Prior NRC Approval

   A risk assessment of the change must demonstrate that the acceptance criteria below are met. The risk assessment approach, methods, and data shall be acceptable to the NRC and shall be appropriate for the nature and scope of the change being evaluated; be based on the as-built, as-operated, and maintained plant; and reflect the operating experience at the plant. Acceptable methods to assess the risk of the change may include methods that have been used in the peer-reviewed fire PRA model, methods that have been approved by NRC through a plant-specific license amendment or NRC approval of generic methods specifically for use in NFPA 805 risk assessments, or methods that have been demonstrated to bound the risk impact.

   a. Prior NRC review and approval is not required for changes that clearly result in a decrease in risk. The proposed change must also be consistent with the defense-in-depth philosophy and must maintain sufficient safety margins. The change may be implemented following completion of the plant change evaluation.

   b. Prior NRC review and approval is not required for individual changes that result in a risk increase less than 1×10-7/year (yr) for CDF and less than 1×10-8/yr for LERF. The proposed change must also be consistent with the defense-in-depth philosophy and must maintain sufficient safety margins. The change may be implemented following completion of the plant change evaluation.

2. Other Changes that May Be Made Without Prior NRC Approval

   a. Changes to NFPA 805, Chapter 3, Fundamental Fire Protection Program.

      Prior NRC review and approval are not required for changes to the NFPA 805, Chapter 3, fundamental fire protection program elements and design requirements for which an engineering evaluation demonstrates that the alternative to the Chapter 3 element is functionally equivalent or adequate for the hazard. The licensee may use an engineering evaluation to demonstrate that a change to an NFPA 805, Chapter 3 element is functionally equivalent to the corresponding technical requirement. A qualified fire protection engineer shall perform the engineering evaluation and conclude that the change has not affected the functionality of the component, system, procedure, or physical arrangement, using a relevant technical requirement or standard.

      The licensee may use an engineering evaluation to demonstrate that changes to certain NFPA 805, Chapter 3 elements are acceptable because the alternative is "adequate for the hazard." Prior NRC review and approval would not be required for alternatives to four specific sections of NFPA 805, Chapter 3, for which an engineering evaluation demonstrates that the alternative to the Chapter 3 element is adequate for the hazard. A qualified fire protection engineer shall perform the engineering evaluation and conclude that the change has not affected the functionality of the component, system, procedure, or physical arrangement, using a relevant technical requirement or standard. The four specific sections of NFPA 805, Chapter 3, are as follows:

      - "Fire Alarm and Detection Systems" (Section 3.8);
      - "Automatic and Manual Water-Based Fire Suppression Systems" (Section 3.9);
      - "Gaseous Fire Suppression Systems" (Section 3.10); and,
      - "Passive Fire Protection Features" (Section 3.11).

      (This License Condition does not apply to any demonstration of equivalency under Section 1.7 of NFPA 805.)

   b. Fire Protection Program Changes that Have No More than Minimal Risk Impact

      Prior NRC review and approval are not required for changes to the licensee's fire protection program that have been demonstrated to have no more than a minimal risk impact. The licensee may use its screening process as approved in the NRC safety evaluation report dated September 8, 2016 to determine that certain fire protection program changes meet the minimal criterion. The licensee shall ensure that fire protection defense-in-depth and safety margins are maintained when changes are made to the fire protection program.

3. Transition License Conditions

   a. Before achieving full compliance with 10 CFR 50.48(c), as specified by 3.b and 3.c below, risk-informed changes to the licensee's fire protection program may not be made without prior NRC review and approval unless the change has been demonstrated to have no more than a minimal risk impact, as described in 2.b above.

   b. The licensee shall implement the modifications to its facility as described in Attachment S, Table S-2 "Plant Modifications Committed," of NextEra Energy Point Beach letter NRC-2016-0013 to complete the transition to full compliance with 10 CFR 50.48(c) no later than prior to startup from the second refueling outage (for each unit) after receipt of the license amendment. The licensee shall maintain appropriate compensatory measures in place until completion of these modifications.

   c. The licensee shall implement the items in Attachment S, Table S-3, "Implementation Items," of NextEra Energy Point Beach letter NRC-2016-0021, with the exception of items noted below, within 12 months after NRC approval unless that falls within a scheduled outage window; then in that case, completion will occur 60 days after the startup from that scheduled outage.

      i. Implementation item 120 is an exception as the industry guidance is under review by the NRC and the final resolution will occur 12 months after the guidance is available unless that falls within a scheduled outage window; then in that case, completion will occur 60 days after startup from that scheduled outage.

      ii. Implementation items 142 and 150 are exceptions because they are associated with completion of committed modifications identified in LAR Attachment S, Table S-2 and will not be completed until 3 months following the last refueling outage identified in item 3.b above.

G. Secondary Water Chemistry Monitoring Program

NextEra Energy Point Beach shall implement a secondary water chemistry monitoring program to inhibit steam generator tube degradation. This program shall include:

1. Identification of a sampling schedule for the critical parameters and control points for these parameters;
2. Identification of the procedures used to quantify parameters that are critical to control points;
3 Identification of process sampling points;
4. Procedure for the recording and management of data;
5. Procedures defining corrective actions for off control point chemistry condition; and
6. A procedure for identifying the authority responsible for the interpretation of the data, and the sequence and timing of administrative events required to initiate corrective action.

H. The licensee is authorized to repair Unit 1 steam generators by replacement of major components. Repairs shall be conducted in accordance with the licensee's commitments identified in the Commission approved Point Beach Nuclear Plant Unit No. 1 Steam Generator Repair Report dated August 9, 1982 and revised March 1, 1983 and additional commitments identified in the staff's related safety evaluation.

I. Deleted

J. Deleted

K. All capsules in the reactor vessel that are removed and tested must meet the test procedures and reporting requirements of American Society for Testing and Materials (ASTM) E 185-82 to the extent practicable for the configuration of the specimens in the capsule. Any changes to the capsule withdrawal schedule, including spare capsules, must be approved by the NRC prior to implementation. All capsules placed in storage must be maintained for future insertion. Any changes to storage requirements must be approved by the NRC, as required by 10 CFR Part 50, Appendix H.

L. Mitigation Strategy

Strategies shall be developed and maintained for addressing large fires and explosions that include the following key areas:

1. Fire fighting response strategy with the following elements:

   a. Pre-defined coordinated fire response strategy and guidance
   b. Assessment of mutual aid fire fighting assets
   c. Designated staging areas for equipment and materials
   d. Command and control
   e. Training of response personnel

2. Operations to mitigate fuel damage considering the following:

   a. Protection and use of personnel assets
   b. Communications
   c. Minimizing fire spread
   d. Procedures for implementing integrated fire response strategy
   e. Identification of readily-available pre-staged equipment
   f. Training on integrated fire response strategy
   g. Spent fuel pool mitigation measures

3. Actions to minimize release to include consideration of:

   a. Water spray scrubbing
   b. Dose to onsite responders

M. Additional Conditions

The additional conditions contained in Appendix C, as revised through Amendment No. 241, are hereby incorporated into this license. NextEra Energy Point Beach shall operate the facility in accordance with the additional conditions.

5. The issuance of this renewed operating license is without prejudice to subsequent licensing action which may be taken by the Commission with regard to the ongoing rulemaking hearing on the Interim Acceptance Criteria for Emergency Core Cooling Systems (Docket No. RM 50-1).

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

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed By

R. W. Borchardt, Deputy Director
Office of Nuclear Reactor Regulation

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

Date of Issuance: December 22, 2005

D. Physical Protection

NextEra Energy Point Beach shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans, including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contain Safeguards Information protected under 10 CFR 73.21, is entitled: "Point Beach Nuclear Plant Physical Security Plan, (Revision 4)," submitted by letter dated May 10, 2006. NextEra Energy Point Beach, LLC shall fully implement and maintain in effect all provisions of the Commission-approved Point Beach Nuclear Plant Cyber Security Plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The NextEra Energy Point Beach CSP was approved by License Amendment No. 247 as supplemented by a change approved by License Amendment No. 251 and License Amendment No. 256.

E. Adoption of 10 CFR 50.69, "Risk-Informed categorization and treatment of structures, systems, and components for nuclear power plants"

1. NextEra Energy Point Beach is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 structures, systems, and components (SSCs) using: Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events, including internal flooding, and internal fire; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (AN0-2) 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 License Amendment No. 265 dated November 26, 2018.

2. Prior to implementation of the provisions of 10 CFR 50.69, NextEra Energy Point Beach shall complete the items below:

   a. Item A in Attachment 1, List of Categorization Prerequisites, to NextEra Energy Point Beach letter NRC 2017-0043, "License Amendment Request 287, Application to Adopt 10 CFR 50.69, 'Risk-Informed Categorization and Treatment of Structures, Systems, and Components (SSCs) for Nuclear Power Plants,' " dated August 31, 2017; and

   b. Attachment 1, Point Beach 10 CFR 50.69 PRA Implementation Items, in NextEra Energy Point Beach letter NRC-2018-0044, "Supplement to Response to Request for Additional Information Regarding License Amendment Request 287, Application to Adopt 10 CFR 50.69, 'Risk informed Categorization and Treatment of Structures, System, and Components (SSCs) for Nuclear Power Plants,' " dated September 28, 2018.

3. 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).

F. NextEra Energy Point Beach Unit 2 shall implement and maintain in effect all provisions of the approved fire protection program that comply with 10 CFR 50.48(a) and 10 CFR 50.48(c), as specified in the license amendment request dated June 26, 2013, and supplements dated September 16, 2013, July 29, 2014, August 28, 2014, September 25, 2014, November 14, 2014, December 19, 2014, January 16, 2015, May 12, 2015, August 26, 2015, February 22, 2016, April 07, 2016, and May 3, 2016 and as approved in the safety evaluation report dated September 8, 2016. Except where NRC approval for changes or deviations is required by 10 CFR 50.48(c), and provided no other regulation, technical specification, license condition or requirement would require prior NRC approval, the licensee may make changes to the fire protection program without prior approval of the Commission if those changes satisfy the provisions set forth in 10 CFR 50.48(a) and 10 CFR 50.48(c), the change does not require a change to a technical specification or license condition, and the criteria listed below are satisfied.

1. Risk-Informed Changes that May Be Made Without Prior NRC Approval

   A risk assessment of the change must demonstrate that the acceptance criteria below are met. The risk assessment approach, methods, and data shall be acceptable to the NRC and shall be appropriate for the nature and scope of the change being evaluated; be based on the as-built, as-operated, and maintained plant; and reflect the operating experience at the plant. Acceptable methods to assess the risk of the change may include methods that have been used in the peer-reviewed fire PRA model, methods that have been approved by NRC through a plant-specific license amendment or NRC approval of generic methods specifically for use in NFPA 805 risk assessments, or methods that have been demonstrated to bound the risk impact

   a. Prior NRC review and approval is not required for changes that clearly result in a decrease in risk. The proposed change must also be consistent with the defense-in-depth philosophy and must maintain sufficient safety margins. The change may be implemented following completion of the plant change evaluation.

   b. Prior NRC review and approval is not required for individual changes that result in a risk increase less than 1×10-7/year (yr) for CDF and less than 1×10-8/yr for LERF. The proposed change must also be consistent with the defense-in-depth philosophy and must maintain sufficient safety margins. The change may be implemented following completion of the plant change evaluation.

2. Other Changes that May Be Made Without Prior NRC Approval

    a. Changes to NFPA 805, Chapter 3, Fundamental Fire Protection Program.

    Prior NRC review and approval are not required for changes to the NFPA 805, Chapter 3, fundamental fire protection program elements and design requirements for which an engineering evaluation demonstrates that the alternative to the Chapter 3 element is functionally equivalent or adequate for the hazard. The licensee may use an engineering evaluation to demonstrate that a change to an NFPA 805, Chapter 3 element is functionally equivalent to the corresponding technical requirement. A qualified fire protection engineer shall perform the engineering evaluation and conclude that the change has not affected the functionality of the component, system, procedure, or physical arrangement, using a relevant technical requirement or standard.

    The licensee may use an engineering evaluation to demonstrate that changes to certain NFPA 805, Chapter 3 elements are acceptable because the alternative is "adequate for the hazard." Prior NRC review and approval would not be required for alternatives to four specific sections of NFPA 805, Chapter 3, for which an engineering evaluation demonstrates that the alternative to the Chapter 3 element is adequate for the hazard. A qualified fire protection engineer shall perform the engineering evaluation and conclude that the change has not affected the functionality of the component, system, procedure, or physical arrangement, using a relevant technical requirement or standard. The four specific sections of NFPA 805, Chapter 3, are as follows:

    - "Fire Alarm and Detection Systems" (Section 3.8);
    - "Automatic and Manual Water-Based Fire Suppression Systems" (Section 3.9);
    - "Gaseous Fire Suppression Systems" (Section 3.10); and,
    - "Passive Fire Protection Features" (Section 3.11).

    (This License Condition does not apply to any demonstration of equivalency under Section 1.7 of NFPA 805.)

    b. Fire Protection Program Changes that Have No More than Minimal Risk Impact

    Prior NRC review and approval are not required for changes to the licensee's fire protection program that have been demonstrated to have no more than a minimal risk impact. The licensee may use its screening process as approved in the NRC safety evaluation report dated September 8, 2016 to determine that certain fire protection program changes meet the minimal criterion. The licensee shall ensure that fire protection defense-in-depth and safety margins are maintained when changes are made to the fire protection program.

3. Transition License Conditions

   a. Before achieving full compliance with 10 CFR 50.48(c), as specified by 3.b and 3.c below, risk-informed changes to the licensee's fire protection program may not be made without prior NRC review and approval unless the change has been demonstrated to have no more than a minimal risk impact, as described in 2.b above.

   b. The licensee shall implement the modifications to its facility as described in Attachment S, Table S-2 "Plant Modifications Committed," of NextEra Energy Point Beach letter NRC-2016-0013 to complete the transition to full compliance with 10 CFR 50.48(c) no later than prior to startup from the second refueling outage (for each unit) after receipt of the license amendment. The licensee shall maintain appropriate compensatory measures in place until completion of these modifications.

   c. The licensee shall implement the items in Attachment S, Table S-3, "Implementation Items," of NextEra Energy Point Beach letter NRC-2016-0021, with the exception of items noted below, within 12 months after NRC approval unless that falls within a scheduled outage window; then in that case, completion will occur 60 days after the startup from that scheduled outage.

      i. Implementation item 120 is an exception as the industry guidance is under review by the NRC and the final resolution will occur 12 months after the guidance is available unless that falls within a scheduled outage window; then in that case, completion will occur 60 days after startup from that scheduled outage.

      ii. Implementation items 142 and 150 are exceptions because they are associated with completion of committed modifications identified in LAR Attachment S, Table S-2 and will not be completed until 3 months following the last refueling outage identified in item 3.b above.

G. Secondary Water Chemistry Monitoring Program

NextEra Energy Point Beach shall implement a secondary water chemistry monitoring program to inhibit steam generator tube degradation. This program shall include:

1. Identification of a sampling schedule for the critical parameters and control points for these parameters;
2. Identification of the procedures used to quantify parameters that are critical to control points;
3. Identification of process sampling points;
4. Procedure for the recording and management of data;
5. Procedures defining corrective actions for off control point chemistry condition; and
6. A procedure for identifying the authority responsible for the interpretation of the data, and the sequence and timing of administrative events required to initiate corrective action.

H. Deleted

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J. All capsules in the reactor vessel that are removed and tested must meet the test procedures and reporting requirements of American Society for Testing and Materials (ASTM) E 185-82 to the extent practicable for the configuration of the specimens in the capsule. Any changes to the capsule withdrawal schedule, including spare capsules, must be approved by the NRC prior to implementation. All capsules placed in storage must be maintained for future insertion. Any changes to storage requirements must be approved by the NRC, as required by 10 CFR Part 50, Appendix H.

K. Mitigation Strategy

Strategies shall be developed and maintained for addressing large fires and explosions that include the following key areas:

1. Fire fighting response strategy with the following elements:

   a. Pre-defined coordinated fire response strategy and guidance
   b. Assessment of mutual aid fire fighting assets
   c. Designated staging areas for equipment and materials
   d. Command and control
   e. Training of response personnel

2. Operations to mitigate fuel damage considering the following:

   a. Protection and use of personnel assets
   b. Communications
   c. Minimizing fire spread
   d. Procedures for implementing integrated fire response strategy
   e. Identification of readily-available pre-staged equipment
   f. Training on integrated fire response strategy
   g. Spent fuel pool mitigation measures

3. Actions to minimize release to include consideration of:

   a. Water spray scrubbing
   b. Dose to onsite responders

L. Additional Conditions

The additional conditions contained in Appendix C, as revised through Amendment No. 245, are hereby incorporated into this license. NextEra Energy Point Beach shall operate the facility in accordance with the additional conditions.

5. The issuance of this renewed operating license is without prejudice to subsequent licensing action which may be taken by the Commission with regard to the ongoing rulemaking hearing on the Interim Acceptance Criteria for Emergency Core Cooling Systems (Docket No. RM 50-1).

6. This renewed operating license is effective as of the date of issuance, and shall expire at midnight on March 8, 2033.

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed By

R. W. Borchardt, Deputy Director
Office of Nuclear Reactor Regulation

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

Date of Issuance: December 22, 2005

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 262 and 265

TO RENEWED FACILITY OPERATING LICENSE NOS. DPR-24 AND DPR-27

NEXTERA ENERGY POINT BEACH, LLC

POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-266 AND 50-301

1.0 INTRODUCTION

By letter dated August 31, 2017 (Reference 1), as supplemented by letters dated October 26, 2017 (Reference 2), August 10, 2018 (Reference 3), and September 28, 2018 (Reference 4), NextEra Energy Point Beach, LLC (NextEra, the licensee) submitted a license amendment request (LAR) for the Point Beach Nuclear Plant, Units 1 and 2 (PBNP). The licensee proposed to add a new license condition to the Renewed Facility Operating Licenses to allow the implementation of Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.69, "Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors." The provisions of 10 CFR 50.69 allow adjustment of the scope of structures, systems, and components (SSCs) subject to special treatment requirements (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation) based on a method of categorizing SSCs according to their safety significance.

By email dated June 27, 2018 (Reference 7), the U.S. Nuclear Regulatory Commission (NRC, the Commission) staff requested additional information (RAI) from the licensee. By letters dated August 10, 2018, and September 28, 2018, the licensee responded to the requests. The supplements provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on February 13, 2018 (83 FR 6226).

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, a 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. Probabilistic risk assessments

(PRAs) address credible initiating events by assessing the event frequency. Mitigating system reliability is then assessed, including the potential for common cause failures.

To take advantage of the safety enhancements available through the use of PRA, the NRC promulgated a new regulation, 10 CFR 50.69, in the *Federal Register* on November 22, 2004 (69 FR 68008), which became effective on December 22, 2004. The provisions of 10 CFR 50.69 allow adjustment of the scope of SSCs subject to special treatment requirements. Special treatment refers to those requirements that provide increased assurance beyond normal industry practices that SSCs perform their design-basis functions. For SSCs categorized as low safety-significance, 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 a RI process, adjusts treatment requirements consistent with the relative significance of the SSC, and manages the process over the lifetime of the plant. A 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. Important-to-safety electrical equipment, that is covered within the scope of 10 CFR 50.69 and relied on for mitigating the design basis accident, should continue to have evidence of environmental qualification demonstrating that the equipment can perform its safety function during and following a design basis accident. 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.

2.2 Licensee's Proposed Changes

In its letter dated September 28, 2018 (Reference 4), the licensee proposed to amend its Renewed Facility Operating Licenses by adding the following license condition that would allow for the implementation of 10 CFR 50.69.

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

to assess passive component risk for Class 2 and Class 3 SSCs and their associated supports; and the results of non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards, i.e., seismic margin analysis (SMA) to evaluate seismic risk, and a screening of other external hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009; as specified in License Amendment No. [NUMBER] dated [DATE].

2. Prior to implementation of the provisions of 10 CFR 50.69, NextEra Energy Point Beach shall complete the items below:

   a. Item A in Attachment 1, List of Categorization Prerequisites, to NextEra Energy Point Beach letter NRC 2017-0043, "License Amendment Request 287, Application to Adopt 10 CFR 50.69, 'Risk-Informed Categorization and Treatment of Structures, Systems, and Components (SSCs) for Nuclear Power Plants,'" dated August 31, 2017; and

   b. Attachment 1, Point Beach 10 CFR 50.69 PRA Implementation Items, in NextEra Energy Point Beach letter NRC-2018-0044, "Supplement to Response to Request for Additional Information Regarding License Amendment Request 287, Application to Adopt 10 CFR 50.69, 'Risk informed Categorization and Treatment of Structures, System, and Components (SSCs) for Nuclear Power Plants,'" dated September 28, 2018.

3. 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 a RI method of categorizing SSCs according to their safety significance. Specifically, for SSCs categorized as low safety-significance (LSS), alternative treatment requirements may be implemented in accordance with the regulation. For SSCs determined to be of high safety-significance (HSS), requirements may not be changed.

Paragraph 50.69(c) of 10 CFR 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 LSS functions

RISC-4: Nonsafety-related SSCs that perform LSS 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 that must, at a minimum, model severe accident scenarios resulting from internal initiating events occurring at full power operation. The PRA must be of sufficient quality and level of detail to support the categorization process, and must be subjected to a peer review process assessed against a standard or set of acceptance criteria that is endorsed by the NRC.

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

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

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

[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.

(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

NEI 00-04, Revision 0, "10 CFR 50.69 SSC Categorization Guideline" (Reference 8), describes a process for determining the safety-significance of SSCs and categorizing them into the four RISC categories defined in 10 CFR 50.69. This categorization process is an integrated decision-making process that incorporates risk and traditional engineering insights. NEI 00-04, Revision 0, provides options for licensees implementing different approaches depending on the scope of their PRA models. It also allows the use of non-PRA approaches when PRAs have not been performed to address 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" (Reference 10), such non-PRA-type evaluations will result in more conservative categorization, in that special treatment requirements will not be allowed to be relaxed for SSCs that are relied upon in such evaluations. The degree of 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. It further states that as part of the NRC's review and approval of a licensee's or applicant's application requesting to implement 10 CFR 50.69, the NRC staff intends to impose a license condition that will explicitly address the scope of the PRA and non-PRA methods used in the licensee's categorization approach. If a licensee or applicant wishes to change its categorization approach and the change is outside the bounds of the NRC's license condition (e.g., switch from a seismic margins analysis to a seismic PRA), the licensee or applicant will need to seek NRC approval, via a license amendment, of the implementation of the new approach in their categorization process. RG 1.201 also states that all aspects of NEI 00-04 must be followed to achieve reasonable confidence in the evaluations required by 10 CFR 50.69(c)(1)(iv).

RG 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities" (Reference 11), describes an acceptable approach for determining whether the quality of the PRA, in total or the parts that are used to support an application, is sufficient to provide confidence in the results, such that the PRA can be used in regulatory decision making for light-water reactors. It endorses, with clarifications, the American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) PRA Standard ASME/ANS RA-Sa-2009 ("ASME/ANS 2009 Standard" or "PRA Standard") (Reference 12). 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" (Reference 13), provides guidance on the use of PRA findings and risk insights in support of changes to a plant's licensing basis. This RG provides risk acceptance guidelines for evaluating the results of such evaluations.

3.0 TECHNICAL EVALUATION

3.1 Staff's Method of Review

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 of this safety evaluation (SE). The staff's review and the documentation of that review in this SE uses the framework of NEI 00-04, Revision 0.

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

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.

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 (IDP) 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; however, the licensee provided little detail of the categorization process. Therefore, in RAI 05 (Reference 7), the NRC staff requested the licensee to: (1) summarize the categorization process, (2) provide the order of the sequence of elements or steps that will be performed, (3) explain the difference between preliminary HSS and assigned HSS, and (4) identify which inputs can and which cannot be changed by the IDP from preliminary HSS to LSS.

In response to RAI 05, the licensee provided a flow chart, and Table 1, shown below, that summarized the categorization process. The licensee provided identification of which steps are performed at the component level and which steps are performed at the function level. The licensee explained that the execution sequence of steps/elements of the process does not impact the resulting preliminary categorization because the safety determination of each element of the process is independent of each other.

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

- Defining system boundaries (see Section 3.3 of this SE).

- Defining system functions 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 (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).

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

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

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

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

As 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 (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.

Table 1

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

With relation to the allowable considerations for qualitative criteria, the licensee provided further clarification:

> The assessments of the qualitative considerations are agreed upon by the IDP in accordance with Section 9.2. In some cases, a 50.69 categorization team may provide preliminary assessments of the seven considerations for the IDP's consideration; however, the final assessments of the seven considerations are the direct responsibility of the IDP.
>
> The seven considerations are addressed preliminarily by the 50.69 categorization team for at least the system functions that are not found to be HSS due to any other categorization step. Each of the seven considerations requires a supporting justification for confirming (true response) or not confirming (false response) that consideration. If the 50.69 categorization team determines that one or more of the seven considerations cannot be confirmed, then that function is presented to

the IDP as preliminary HSS. Conversely, if all the seven considerations are confirmed, then the function is presented to the IDP as preliminary LSS.

The System Categorization Document, including the justifications provided for the qualitative considerations, is reviewed by the IDP. The IDP is responsible for reviewing the preliminary assessment to the same level of detail as the 50.69 team (i.e., all considerations for all functions are reviewed). The IDP may confirm the preliminary function risk and associated justification or may direct that it be changed based upon their expert knowledge. Because the Qualitative Criteria are the direct responsibility of the IDP, changes may be made from preliminary HSS to LSS or from preliminary LSS to HSS at the discretion of the IDP. If the IDP determines any of the seven considerations cannot be confirmed (false response) for a function, then the final categorization of that function is HSS.

The NRC staff has evaluated the categorization steps and the associated clarifications provided by the licensee in response to RAI 05 and Table 1, 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.

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

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

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

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

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

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. Section 3.1.1 of the LAR also states that the SSC categorization process documentation will include, among other items, system functions identified and categorized with the associated bases and mapping of components to support function(s).

In response to RAI 05.g (Reference 3), the licensee confirmed that it will follow the guidance in NEI 00-04 that any functions/SSCs that serve as the interface between two or more systems will not be categorized until the categorization of all systems that they support is complete.

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 50.69(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 (high winds, external floods, etc.), and shutdown, 10 CFR 50.69(b)(2) allows, and the NEI 00-04 guidance summarizes, the use of PRA if such PRA models exist, or, in the absence of quantifiable PRA, the use of other methods (e.g., fire-induced vulnerability evaluation, SMA, IPEEE screening, and shutdown safety plan).

As stated in LAR Sections 3.1.1 and 3.2.1 through 3.2.5, as further clarified in the supplement dated October 26, 2017 (Reference 2), the licensee's categorization process uses PRA to assess risks from internal events (including internal flooding) and from fire. For the other three risk hazard groups, the licensee's process uses non-PRA methods for the risk characterization, as follows:

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

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

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

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

Internal Events PRA

The NRC staff's review of the internal events and flooding PRAs was based on the results of the peer review of the internal events and flooding PRAs; the associated facts and observations (F&O) closure review described in LAR Sections 3.2.1 and 3.3, and presented in LAR Attachment 3; and previously docketed information on PRA quality submitted to the NRC in the request for implementation of technical specifications task force (TSTF) traveler TSTF-425, regarding the relocation of surveillance frequencies to a licensee controlled program, dated July 3, 2014 (Reference 5). The last full-scope peer review of the internal events PRA (including internal flooding) was performed in November 2010. A focused-scope peer review of the internal flooding PRA was performed in August 2011, and a focused-scope peer review of the internal events PRA excluding internal flooding, was performed in October 2011 of the updated internal events and flooding PRAs. All peer reviews were performed against the ASME/ANS 2009 Standard and RG 1.200, Revision 2.

An Independent Assessment (IA) F&O closure review, which the NRC staff observed (Reference 6), was performed in July 2017 by an independent team for the internal events, internal flooding, and fire finding-level F&Os. The F&O closure review process is described in Appendix X (Reference 17) to the guidance in NEI 05-04 (Reference 18), NEI 07-12 (Reference 19), and NEI 12-13 (Reference 20), as the process for "Close-Out of Facts and Observations." The staff accepted, with conditions, a final version of Appendix X to NEI 05-04, NEI 07-12, and NEI 12-13 by letter dated May 3, 2017 (Reference 21).

The NRC staff requested confirmation in RAI 01 that the IA F&O closure process was performed in accordance with NRC-accepted guidance. In response to RAI 01, the licensee provided the necessary details to confirm that the closure review met the accepted guidance. The licensee stated that on a follow-up closure review, the IA team was provided a written assessment of the findings and determined whether any resolutions met the ASME/ANS 2009 Standard (Reference 12) definition of PRA upgrade. The licensee stated that all of the resolutions were assessed as maintenance updates and concurred upon by the IA team. The licensee also confirmed that it met the criteria for selecting the IA team members, ensured that the assessment was performed to Capability Category II requirements, and that the review encompassed all finding level F&Os.

In LAR, Attachment 3, the licensee submitted all of the internal event and internal flooding peer review F&Os that remained open from the IA F&O closure review. For each F&O, the licensee provided a disposition for this application. Additionally, the NRC staff used in its review of the F&Os previously docketed information submitted as part of the licensee's request for implementation of TSTF-425, regarding the relocation of surveillance frequencies to a licensee controlled program (Reference 5).

The NRC staff reviewed the licensee's resolution of all of the peer review findings presented in the LAR and the TSTF-425 LAR and assessed the potential impact of the findings on the 10 CFR 50.69 categorization. The NRC staff requested additional information to clarify the licensee's disposition for some of the findings and also requested additional information to clarify how the licensee addressed other modeling concerns, as described in the following paragraphs.

In the disposition to F&O IE-A1-01 presented in the TSTF-425 LAR (Reference 5), the licensee stated that a number of special initiators associated with the 4160 VAC Vital Switchgear bus were not included because they were not considered significant contributors to CDF. In RAI 02.a (Reference 7), the NRC staff requested further justification that exclusion of the cited initiating events related to the 4160 VAC Vital Switchgear bus has no impact on 10 CFR 50.69 categorization. In its letter dated September 28, 2018 (Reference 4), the licensee proposed implementation item ii to add the loss of the 4160 VAC bus as a special initiator in the PRA model prior to implementation of the 10 CFR 50.69 categorization process.

In the disposition to F&O AS-B6-01 and F&O SY-A21-01 presented in the TSTF-425 LAR (Reference 5), the licensee stated that modeling of the Emergency Diesel Generator (EDG) load management was excluded from the PRA because the licensee estimated a low likelihood that the EDG load management would be needed. In RAI 02.b (Reference 7), the NRC staff requested further justification that exclusion of the load management actions has no impact on 10 CFR 50.69 categorization. In its letter dated September 28, 2018 (Reference 4), the licensee proposed implementation item iii to add the new failure mode associated with the EDG load management to the PRA model prior to implementation of the 10 CFR 50.69 categorization process.

In the disposition to F&O AS-B7-01 presented in the TSTF-425 LAR (Reference 5), the licensee stated that recovery of loss-of-offsite power (LOOP) events is only credited for station blackout (SBO) scenarios and that the direct current batteries are conservatively assumed to fail at time zero. The NRC staff noted in RAI 02.c (Reference 7), that conservative modeling can skew the plant's risk profile and impact the SSC importance values determined as part of the 10 CFR 50.69 categorization. Therefore, the NRC staff requested justification that the conservative modeling would have no impact on the 10 CFR 50.69 categorization. In its letter dated September 28, 2018 (Reference 4), the licensee proposed implementation item iv to revise the treatment of power recovery after LOOP events and battery modeling in the PRA model to be more realistic prior to implementation of the 10 CFR 50.69 categorization process.

In F&O HR-D1-01 regarding human reliability analysis (HRA) pre-initiator screening, the licensee stated that no further PRA model changes are required, but that prior to the implementation of the 10 CFR 50.69 categorization process, this finding will either be closed or a sensitivity study will be performed to determine the impact of the F&O on the categorization. In RAI 02.d (Reference 7), the NRC staff requested explanation for why the F&O could not be closed and to justify why this F&O has no impact on the 10 CFR 50.69 categorization. In response to RAI 02.d (Reference 3), the licensee explained that it is open because pertinent documentation could not be provided to the IA closure team. In its letter dated September 28, 2018 (Reference 4), the licensee proposed implementation item v to resolve and close F&O HR-D1-01 using an NRC-accepted process (e.g., full-scope peer review, focused-scope peer review, or F&O closure review) prior to implementation of the 10 CFR 50.69 categorization process.

The F&O IFQU-A6-01 presented in LAR, Attachment 3, found that human failure events (HFEs) used in the internal flooding PRA were adjusted by the licensee without an adequate basis. Therefore, in RAI 02.e (Reference 7), the NRC staff requested justification that scenario-specific internal flooding HFEs were developed for the internal flooding PRA and that factors used to account for flooding scenario stress were appropriate. In response to RAI 02.e (Reference 3), the licensee explained that the internal flooding PRA documentation will be updated to justify that the human error probabilities (HEPs) used in the model resolve this finding. In its letter dated September 28, 2018 (Reference 4), the licensee proposed implementation item vi to resolve and close F&O IFQU-A6-01 using an NRC-accepted process (e.g., full-scope peer review, focused-scope peer review, or F&O closure review) prior to implementation of the 10 CFR 50.69 categorization process.

In RAI 03 (Reference 7), the NRC staff requested confirmation that any PRA modeling changes associated with resolutions for F&Os discussed in RAIs 01 and RAI 02 or any modeling uncertainties discussed in RAI 08 do not constitute a "PRA upgrade" as defined by the ASME/ANS 2009 Standard. In response to RAI 03 (Reference 3), the licensee stated that there have been no PRA upgrades for any of the closed findings, and this was confirmed by the independent review team. In its letter dated September 28, 2018 (Reference 4), the licensee proposed implementation item x to independently review all changes performed to the PRA to address the implementation items to determine whether the resolution of those items in the model constitutes an upgrade or update. The implementation item further states that if a PRA upgrade is identified, a focused-scope peer review will be performed for that upgrade, and any resulting F&Os will be resolved in the PRA to meet Capability Category II.

In RAI 10 (Reference 7), the NRC staff requested clarification of: (1) whether the internal events and fire PRA models credited the Westinghouse Generation III reactor coolant pump (RCP) seals; (2) whether the current guidance in tropical report (TR) of the Pressurized-Water Reactor Owners Group (PWROG)-14001-P, Revision 1 (Reference 22), was followed and the limitations and conditions outlined in the NRC SE for use of that guidance (Reference 23) were met, and (3) whether addition of the RCP seals in the internal events and fire PRA models meets the definition of a PRA upgrade as defined by the ASME/ANS 2009 Standard. In response to RAI 10 (Reference 3), the licensee explained that the Westinghouse Generation III RCP seals are modeled in the fire PRA and will be modeled in the internal events PRA that will be used for 10 CFR 50.69 categorization, consistent with the guidance in PWROG-14001-P, Revision 1. The licensee also stated that there will be no exceptions to the limitations and conditions prescribed in the NRC SE for modeling the RCP seals. In its letter dated September 28, 2018 (Reference 4), the licensee proposed implementation item ix to update both the internal events and fire PRA models to credit the Westinghouse Generation III RCP seals using the guidance from PWROG-14001-P, Revision 1, and the limitations and conditions in the associated NRC SE. Implementation item ix also states that the additional failure contribution of the Westinghouse RCP shutdown seal bypass failure mode will be added to the PRA models, consistent with the limitations and conditions in the NRC SE for PWROG-140001-P, Revision 1. Additionally, in response to RAI 10, the licensee explained that the updated RCP seal modeling will not meet the definition of a PRA upgrade because: (1) the modeling does not represent a new method as it is simply an expansion of the current peer-reviewed RCP seal model with additional seal and human failure probabilities and different seal leakage rates, (2) there is no change in the model scope because the equipment, dependencies, and types of accident sequences remain the same, and (3) there is no change in PRA modeling capability given that the model can still evaluate the risk associated with station blackout and total loss of cooling events related to RCP seal failures.

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

Fire PRA

The NRC staff reviewed the results of the peer review of the fire PRA and associated F&O closure review described in LAR Sections 3.2.2 and 3.3, and presented in LAR, Attachment 3. The licensee's fire PRA was subject to a full-scope industry peer review in June 2011, a focused-scope peer review on the fire scenario selection element of the PRA standard in May 2013, and a focused-scope peer review on the Fire Quantification element of the PRA standard in June 2013. The licensee stated in its TSTF-425 LAR for a licensee controlled risk-informed surveillance frequency program (Reference 5), that all fire PRA peer reviews were performed against the ASME/ANS 2009 Standard in conformance with RG 1.200, Revision 2.

As discussed above, an IA F&O closure review was performed in July 2017 by an independent assessment team, which covered the fire finding-level F&Os. The IA F&O closure review is discussed above in the review of the internal events PRA. In LAR, Attachment 3, the licensee submitted all of the fire F&Os that remained open from the IA F&O closure review. For each F&O, the licensee provided a disposition for this application.

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

The F&O PRM-B2-01 identified two open internal events F&Os (AS-B6-01 and SY-A21-01) that can impact the fire PRA. These two F&Os are the subject of RAI 02.b discussed above in the internal events PRA quality subsection. In RAI 02.f (Reference 7), the NRC staff requested justification for not making this internal events related update in the fire PRA prior to implementing the 10 CFR 50.69 categorization. In its letter dated September 28, 2018 (Reference 4), the licensee proposed implementation item iii to model EDG load management actions in the fire PRA, as well as in the internal events PRA, to resolve fire PRA F&O PRM-B2-01 prior to the implementation of the 10 CFR 50.69 categorization process.

The F&O HRA-B2-01 found that credit for graphically distinct procedural steps was taken for all HRA events rather than only for steps to which this credit applies. In the F&O disposition, the licensee stated that this observation only applied to about 10 percent of HEPs and, therefore, had a minimal impact on the application. In RAI 02.g (Reference 7), the NRC staff requested that the licensee justify that the unfounded credit has no impact on the application or propose a mechanism that ensures that F&O HRA-B2-01 will be resolved prior to implementing the 10 CFR 50.69 categorization process. In its letter dated September 28, 2018 (Reference 4), the licensee proposed implementation item vii to remove credit for graphically distinct procedural steps in the fire PRA HEPs and update the fire HEP dependency analysis prior to implementation of the 10 CFR 50.69 categorization process.

The F&O FQ-A1-01 found FRANX database basic events mapped to scenarios, components, and cables not found in the CAFTA model. The licensee's F&O disposition stated that the "mapping table" needed to be updated in the documentation and "should not" affect the application. However, the NRC staff noted that in the disposition to F&O FQ-A1-01 in the licensee's request to transition to National Fire Protection Association (NFPA) 805 (Reference 14), the licensee indicated that six Main Control Room failure events associated with this issue were still excluded from the fire PRA model. Therefore, in RAI 02.h (Reference 7), the NRC staff requested justification that the excluded events have no impact on the 10 CFR 50.69 categorization or that the licensee propose a mechanism that ensures that this F&O will be resolved prior to implementing the 10 CFR 50.69 categorization process. In its letter dated September 28, 2018 (Reference 4), the licensee proposed implementation item viii to review and update the basic event mapping tables and incorporate the updated mapping into the fire PRA prior to implementation of the 10 CFR 50.69 categorization process.

As discussed above for the internal events PRA, the licensee stated in response to RAI 03 (Reference 3), that there have been no PRA upgrades for any of the closed findings and this was confirmed by the independent review team. In its letter dated September 28, 2018 (Reference 4), the licensee proposed implementation item x to independently review all changes performed to the PRA to address the implementation items to determine whether the resolution of those items in the PRA model constitutes a PRA upgrade or update. The implementation item further states that if a PRA upgrade is identified, a focused-scope peer review will be performed for that upgrade and any resulting F&Os will be resolved in the PRA to meet Capability Category II.

As previously discussed in the SE Section above on the internal events PRA, in its letter dated September 28, 2018 (Reference 4), the licensee proposed implementation item ix to update both the internal events and fire PRA models to credit the Westinghouse Generation III RCP seals using the guidance from PWROG-14001-P, Revision 1, and the limitations and conditions in the associated NRC staff SE. Implementation item ix also states that the additional failure contribution of the Westinghouse RCP shutdown seal bypass failure mode will be added to the PRA models, consistent with the limitations and conditions in the NRC staff SE for PWROG-140001-P, Revision 1.

Paragraph 50.69(c)(1)(i) of 10 CFR requires, in part, that any plant-specific PRA used in the categorization must be of sufficient quality and level of detail to support the categorization process and must be subjected to a peer review process assessed against a standard that is endorsed by the NRC. RG 1.200 provides guidance for determining the technical adequacy of a fire PRA by comparing the PRA to the relevant parts of the ASME/ANS 2009 Standard using a peer review process. Based on its review, the NRC staff finds that the licensee has followed the guidance in RG 1.200 and submitted the results of the peer review and, therefore, meets the requirement in 10 CFR 50.69(b)(2)(iii). The LAR states that the fire PRA model was developed consistent with guidance in NUREG/CR-6850 (Reference 26) and only uses NRC-approved methods. The NRC staff has reviewed the peer review results and the licensee's resolution of the results. Significant errors and weaknesses with the fire PRA will be resolved with the completion of implementation items iii, vii, viii, ix, and x. Therefore, the NRC staff concludes that the quality and level of detail of the fire PRA, with the completion of the implementation items, is sufficient to support the categorization of SSCs using the process endorsed by the NRC staff in RG 1.201 and, therefore, it meets the requirement in 10 CFR 50.69(c)(1)(i).

3.5.2 Importance Measures and Sensitivity Studies

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

Fussell-Vesely (F-V) and Risk Achievement Worth (RAW) importance measures are obtained for each component and each PRA modeled hazard (i.e., separately for the internal events PRA and for the fire PRA) and the values are compared to specified criteria in NEI 00-04. Components that have internal event importance measure values that exceed the criteria are assigned HSS and cannot be changed by the IDP. Components that have fire event importance measures exceeding the criteria are assigned preliminary HSS. Integrated importance measures over all PRA modeled hazards are calculated per Section 5.6 of NEI 00-04, and components for which these measures exceed the criteria are assigned preliminary HSS.

The guidance in NEI 00-04 specifies sensitivity studies to be conducted for each PRA model. The sensitivity studies are performed to ensure that assumptions associated with these specific 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) and fire PRAs that should be evaluated with a sensitivity study. The licensee used the NRC guidance in NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making" (Reference 24), supplemented with the Electric Power Research Institute (EPRI) TR-1016737, "Treatment of Parameter and Modeling Uncertainty for Probabilistic Risk Assessments" (Reference 25), to identify sources of uncertainty in the internal events PRA.

In LAR, Attachment 6, the licensee provided a list of key assumptions and sources of modeling uncertainty that were reviewed for the internal events (including internal flooding) and fire PRAs and dispositions for each entry. The assessment concluded that, in general, no additional sensitivity analyses were needed to address PRA model-specific assumptions or sources of uncertainty. The NRC staff reviewed this listing and found that two key assumptions and sources of uncertainties might impact the application. In RAI 08 (Reference 7), the NRC staff requested further justification for two of the licensee identified key assumptions and sources of uncertainties, as discussed below.

The licensee identified as a source of uncertainty that the operator actions to control the flow of auxiliary feedwater (AFW) later in an accident sequence are not explicitly modeled in the AFW system fault trees. The licensee explained that it performed a sensitivity analysis evaluating the impact of not controlling AFW flow for the full PRA mission time, and, based on this study, the exclusion of operator action to control AFW flow late in the accident sequence has a "small" impact. In RAI 08.a (Reference 7), the NRC staff requested further justification why this uncertainty has no impact on the 10 CFR 50.69 categorization process. In response to RAI 08.a (Reference 3), the licensee explained that it increased an existing AFW HEP value in the model by 1E-4 to account for this issue, which resulted in a 2E-7 increase in CDF. The NRC staff finds the licensee's response acceptable because the results of the sensitivity study verify the licensee conclusion that the results are not sensitive to the change in this HEP value, and because the sensitivity studies on all HEPs that are to be conducted during the categorization process, as described in Tables 5-2 and 5-3 of NEI 00-04, are used to ensure that the values chosen for all HEPs are not masking the importance of any SSCs.

In RAI 08.b (Reference 7), the NRC staff requested justification that exclusion of modeling expansion joint failures in the fire protection system has no impact on the 10 CFR 50.69 categorization process. In response to RAI 08.b (Reference 3), the licensee stated that the failure rate associated with the fire protection system expansion joints is negligible when compared to other failure modes associated with the system, such as the diesel pump failure to run, which the licensee showed is over three orders of magnitude more likely to occur. Therefore, the NRC staff finds that no additional sensitivity analyses are needed to address this source of PRA modeling uncertainty.

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) and fire PRAs consistent with the guidance in NUREG-1855 and EPRI document TR-1016737 and, therefore, satisfied the NEI 00-04 guidance to identify additional "applicable sensitivity studies."

3.5.3 Non-PRA Methods

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

As described in the LAR, and further clarified in the supplement dated October 26, 2017 (Reference 2), the licensee's categorization process uses the following non-PRA methods:

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

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

Seismic Risk

To assess seismic risk for the 10 CFR 50.69 categorization process, the licensee proposes to use the SMA method. SMA is a screening method that does not quantify CDF. The licensee used the EPRI SMA method described in EPRI NP-6041-SL, Revision 1, "A Methodology for Assessment of Nuclear Power Plant Seismic Margin" (Reference 27), during the IPEEE (NRC, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10 CFR 50.54(f)," Generic Letter 88-20, Supplement 4), dated June 1991 (Reference 28). The SMA method includes the development of the seismic safe shutdown equipment list (SSEL), which contains the components that would be needed during and after a seismic event. The SSEL identifies one preferred and one alternate path capable of achieving and maintaining safe shutdown conditions for at least 72 hours following an earthquake. In response to RAI 04 (Reference 3), as updated in letter dated September 28, 2018 (Reference 4), the licensee proposed implementation item i to perform an evaluation and update the SMA SSEL to reflect the as-built, as-operated plant prior to the implementation of the 10 CFR 50.69 categorization process.

Consistent with NEI 00-04, the licensee's categorization process considers all components in the SSEL as HSS based on seismic risk.

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

Other External Hazards (High Winds, External Floods)

The licensee stated that external hazards were initially evaluated by the licensee during the IPEEE. This hazard category includes all non-seismic external hazards such as high winds, external floods, transportation and nearby facility accidents, and other hazards. The IPEEE external hazard analysis used a progressive screening approach and concluded that all these other hazards are negligible contributors to overall plant risk. Further, the licensee indicated that it had reevaluated these other external hazards using the criteria in the ASME/ANS 2009 Standard and screened all external hazards beyond seismic events. In response to RAI 06.a (Reference 3), which requested additional information on the external event screening, the licensee stated that an evaluation was performed to determine if there are components that participate in screened scenarios and whose failure would result in an unscreened scenario. This step is consistent with the process summarized in Figure 5-6 of NEI 00-04.

As clarified in the LAR supplement dated October 26, 2017 (Reference 2), the licensee screened the extreme winds and tornadoes hazard with the justification that the bounding CDF associated with the hazard is less than 1E-06/year. Given that credit for SSCs participating in screening of a hazard can potentially impact the screening, the NRC staff requested in RAI 06.e (Reference 7), explanation and justification for how the guidance in Figure 5-6 of NEI 00-04 was applied. In response to RAI 06.e (Reference 3), the licensee stated that screening of this hazard was performed consistent with the guidance in NEI 00-04, Figure 5-6 and included evaluation of SSCs participating in screened scenarios to determine if removal of credit of those SSCs would cause the scenario to become unscreened. The licensee presented estimates from the IPEEE of core melt probability contributions of vulnerable SSCs that are unprotected from extreme winds and tornadoes. Based on the licensee's estimates, removal of credit for protection of the SSCs does not cause the hazard to be unscreened except for the EDG exhaust stacks. With regard to the EDG exhaust stacks, the licensee explained that during the IPEEE, the G01 and G02 diesel generator exhaust stacks were modified to accommodate higher wind speeds, and estimated a core melt probability contribution of the EDG exhaust stacks to 2E-7. The licensee stated that this estimate did not credit the two additional EDGs (G03 and G04) that were installed after the IPEEE evaluation, along with the necessary support systems and new housing. The licensee stated that the new EDGs G03 and G04 are not vulnerable to high winds and tornadoes hazards and either of them can provide the power required to address design basis accident mitigation. Given the added redundancy provided by G03 and G04, the licensee concluded that there are no SSCs credited for screening of high winds that, if removed, would cause the scenario to become unscreened. The NRC staff did not review the licensee's numerical risk estimates, however, it concludes that the licensee adequately applied the screening process described in NEI 00-04, Figure 5-6, and, therefore, finds this acceptable.

The licensee stated that the external flooding hazard was screened from consideration in the 10 CFR 50.69 process because events associated with this hazard are bounded by the current licensing basis (CLB) or, in the case of a local intense precipitation (LIP) event, there is "an acceptable method of assuring safe shutdown." Regarding LIP, the licensee stated that "implementing the FLEX[2] strategy provide[s] an acceptable method of assuring safe shutdown." In RAI 06.f (Reference 7), the NRC staff requested specific information about screening the external flooding hazard, given that LIP was found by the licensee not to be bounded by the CLB (Reference 16). In response to RAI 06.f (Reference 3), the licensee stated that there are no SSCs credited in the screening of the LIP flood event. The licensee stated that though the LAR indicated FLEX strategies are relied on for LIP mitigation, the identified flooding vulnerability associated with the EDG exhaust stack has been corrected. The licensee stated that given this correction, all key safety functions remain available in the postulated LIP event. Therefore, according to the licensee's response to the NRC's 10 CFR 50.54(f) request for

2 In order to meet certain NRC regulations and orders (such as NRC Order EA-12-049, "Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events," issued after the Fukushima Dai-ichi accident), the nuclear industry has developed mitigating strategies which include procuring portable equipment to restore or maintain various safety functions during beyond-design-basis conditions and the loss of permanently installed plant equipment. This equipment and procedures are referred to as "FLEX strategies" where FLEX is not an acronym.

information regarding evaluation of external flooding (Reference 15), and the licensee's focused evaluation for a LIP event (Reference 16), and given that the EDG exhaust flooding vulnerability is corrected, all floods are now bounded by the CLB. The licensee concluded that there are no SSCs credited in screening external flooding events.

Because the licensee confirmed that the other external hazards 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 (Reference 9). The guidance in NUMARC 91-06 provides considerations for maintaining DID for the five key safety functions during shutdown, namely, decay heat removal capability, inventory control, power availability, reactivity control, and containment - primary/secondary. The guidance in NUMARC 91-06 specifies that a DID approach should be used with respect to each defined shutdown key safety function. This is accomplished by designating a running and an alternative system/train to accomplish the given key safety function.

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

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

- If the 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 addresses the passive function of active components, such as the pressure/liquid retention function of the body of a motor-operated valve.

In the LAR, the licensee proposed using a categorization method for passive components not cited in NEI 00-04 for passive component categorization but approved by the NRC for ANO-2 (Reference 29). The ANO-2 methodology is a RI safety classification and treatment program for repair/replacement activities for Class 2 and Class 3 pressure-retaining items and their associated supports, 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." The ANO-2 methodology relies on the conditional core damage and large early release probabilities associated with pipe ruptures. Safety-significance is generally measured by the frequency and the consequence of, in this case, pipe ruptures. Treatment requirements (including repair/replacement) only affect the frequency of passive component failure. Categorizing solely based on consequences, which measures the safety-significance of the pipe given that it ruptures, is conservative compared to including the rupture frequency in the categorization. The categorization will not be affected by changes in frequency arising from changes to the treatment. Therefore, the NRC staff finds that the use of the repair/replacement methodology is acceptable and appropriate for passive component categorization of Class 2 and Class 3 SSCs.

In the LAR, the licensee did not specify what class of passive components will be categorized with the ANO-2 methodology. In RAI 09 (Reference 7), the NRC staff requested the licensee to either confirm that only Class 2 and Class 3 SSCs will be categorized using ANO-2 passive methodology or to explain and justify how the methodology will be modified to include Class 1 components. In response to RAI 09 (Reference 3), the licensee stated that the NRC-approved ANO-2 approach for passive categorization will only be applied to Class 2, Class 3, and non-Code class components. The licensee stated that all ASME Code Class 1 SSCs with a pressure retaining function, as well as supports, will be designated HSS for passive categorization and cannot be changed by the IDP.

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

3.5.5 Summary

The NRC staff reviewed the PRA and the non-PRA methods used by the licensee in 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, including internal flooding risk
- Fire PRA to assess fire risk
- SMA to assess seismic risk
- Screening using IPEEE to assess risk from other external hazards (high winds, external floods)
- Shutdown safety management plan to assess shutdown risk

- ANO-2 (see Reference 29) passive categorization method to assess passive component risk for Class 2, Class 3, and non-Code class SSCs and their associated supports

Based on its review of the LAR and the licensee's responses to the NRC staff RAIs, the staff identified certain specific actions necessary to support its conclusion that the proposed program meets the requirements in 10 CFR 50.69 and the guidance in RG 1.201 and NEI 00-04. The licensee proposed the addition of a license condition for the implementation of 10 CFR 50.69. The license condition identifies ten implementation items that shall be completed prior to the implementation of the 10 CFR 50.69 categorization process, nine of which are regarding updates to the internal events and fire PRA, and one of which is associated with categorization:

i. NextEra will perform an evaluation of the as-built, as-operated plant against the SMA SSEL. The SSEL updated to reflect the as-built, as-operated plant will be used in the 10 CFR 50.69 categorization.

ii. The loss of a 4,160 VAC bus will be added to the PRA model as a special initiator to resolve internal events finding F&O IE-A1-01, as indicated in response to RAI 02.a.

iii. A new failure mode associated with EDG load management will be added to the internal events and fire PRA model to resolve internal events findings AS-B6-01 and SY-A21-01 and fire finding PRM-B2-01, as indicated in response to RAI 02.b and 02.f.

iv. The treatment of power recovery after LOOP events and battery modeling in the PRA model will be revised to be more realistic to resolve internal events finding AS-B7-01, as indicated in response to RAI 02.c.

v. F&O HR-D1-01 will be closed by the licensee using an NRC-accepted process (e.g., full-scope peer review, focused-scope peer review, or F&O closure review in accordance with Appendix X accepted by the NRC, with conditions, in letter dated May 3, 2017).

vi. F&O IF-QU-A6-01 will be closed by the licensee using an NRC-accepted process (e.g., full-scope peer review, focused-scope peer review, or F&O closure review in accordance with Appendix X accepted by the NRC, with conditions, in letter dated May 3, 2017).

vii. The HEPs developed for the fire PRA model will be updated to remove the graphically distinct credit in the cognitive portion of the HEP. The dependency analysis will be updated and the fire PRA quantified using these updated Human Error Probabilities, as indicated in response to RAI 02.g.

viii. The basic event mapping tables in the fire PRA will be reviewed and compared to the present basic event mapping associated with each equipment or cable. Those items that are no longer needed will be removed and any incorrect mapping will be updated. The Fire PRA model will be quantified using this updated mapping table, as indicated in response to RAI 02.h.

ix. Update the internal events and fire PRA models to credit the Westinghouse Generation III RCP seals using the guidance from PWROG-14001-P, Revision 1, and the limitations and conditions in the associated NRC's safety evaluation (ADAMS Accession Number ML17200A116), as stated in response to RAI 10. The additional failure contribution of the Westinghouse RCP shutdown seal bypass failure mode will be added to the PRA models, consistent with the limitations and conditions in the NRC safety evaluation for PWROG-140001-P, Revision 1.

x. All changes performed to the PRA to address the above implementation items will be independently reviewed to determine if the resolution of those items in the PRA model constitutes a PRA upgrade. If the review identifies a change is a PRA upgrade, a focused-scope peer review will be performed for that change, and any resulting F&Os will be resolved in the PRA to meet Capability Category II.

Additionally, the license condition states, in part, that prior NRC approval is required for a change to the categorization process that is specified in the license amendment and its supplements. The NRC staff's evaluation of the proposed license condition is in Section 3.12 of this SE.

3.6 DID (NEI 00-04, Section 6)

Paragraph 50.69(c)(1)(iii) of 10 CFR requires that the process used for categorizing SSCs must maintain DID. NEI 00-04, Section 6, provides guidance on assessment of DID. In Section 3.1.1 of the LAR, the licensee states 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 likelihood of the design-basis internal event initiating event and the number of redundant and diverse trains nominally available to mitigate the initiating event. The likelihood of the initiating events is binned and, for different likelihood bins, HSS is assigned if fewer than the indicated number of mitigating trains are nominally available. Section 6 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.

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

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 that 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.

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

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

In LAR Section 3.1.1, the licensee stated that if any component is identified as HSS from either the integrated PRA component safety-significance assessment (Section 5 of NEI 00-04), or the DID assessment (Section 6 of NEI 00-04), the associated system function(s) would be identified as HSS. Once a system function is identified as HSS then all the components that support that function are categorized as preliminary HSS. In RAI 05.d (Reference 7), the NRC staff requested the licensee to clarify whether all aspects identified in Sections 5 and 6 of NEI 00-04, including if any components identified as HSS through Sections 5.3 to 5.5 of NEI 00-04 (dedicated to seismic, external hazards, or shutdown risk) will drive the system functions to be categorized as HSS. In response to RAI 05.d (Reference 3), the licensee explained that the safety-significance of functions will be categorized as preliminary HSS only if it is supported by a component determined to be HSS from a PRA-based assessment (i.e., for PBNP, internal events PRA and integrated PRA importance measures described in Section 5.6 of NEI 00-04). Components that are identified as HSS from using the non-PRA approaches (SMA, shutdown risk, other external hazards) will not drive the system function(s) they support to be assigned HSS. The licensee explained that non-PRA-based assessments result in the default categorization of any components associated with the safe shutdown success paths defined in those deterministic assessments to be HSS regardless of its risk significance.

The 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 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. LAR Section 3.1.1 states 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. In addition, Section 3.1.1 states 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).

In response to RAI 05 (Reference 3), the licensee explained that the IDP's authority to change component categorization from preliminary HSS to LSS is limited. The licensee summarized these limitations in Table 1 of the response to RAI 05. Consistent with the guidance in NEI 00-04, components found to be HSS from the following aspects of the process cannot be re-categorized by the IDP:

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

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

In RAI 05.f (Reference 7), the NRC staff requested that the licensee explain how the qualitative criteria in Section 9.2 of NEI 00-04 will be used to determine the safety-significance of the system functions. In response to RAI 05.f (Reference 3), the licensee stated that the final assessment of the qualitative criteria is the direct responsibility of the IDP and that if the IDP determines that any one of them cannot be confirmed (false response) for a function, then the final categorization of that function is HSS. The NRC staff finds that the licensee's proposed use of the seven qualitative questions in the 10 CFR 50.69 categorization process is consistent with the guidance in NEI 00-04 and, therefore, acceptable.

The IDP may change the categorization of a component from LSS to HSS based on its assessment and decision-making. As outlined in NEI 00-04, Section 10.2, and confirmed by the licensee in response to RAI 05 (Reference 3), the IDP may re-categorize components supporting an HSS function from HSS to LSS only if: (1) a credible failure of the component would not preclude the fulfillment of the HSS function and (2) the component was not categorized as HSS based on internal events PRA, integrated PRA component risk, SMA, other external hazards, shutdown, DID, or passive categorization. The licensee also explained that NEI 00-04, Section 4.0, discusses additional functions that may be identified (e.g., fill and drain) to group and consider potentially LSS components that may have been initially associated with an HSS function but that do not support the critical attributes of that HSS function.

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

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

Paragraph 50.69(c)(1)(ii) of 10 CFR requires, in part, that all aspects of the integrated, systematic process used to characterize SSC importance must reasonably reflect the current plant configuration and operating practices and applicable plant and industry operating experience. 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 several weaknesses or errors in the PRA will be addressed, as stated in the implementation items prior to implementation of the 10 CFR 50.69 categorization. The results of the review of the current PRA are reported in Section 3.5 of this SE.

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

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

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

In addition, LAR, Attachment 1 (List of Categorization Prerequisites), states that the licensee will establish procedures prior to the use of the categorization process that will contain the following elements: (1) IDP member qualification requirements, (2) qualitative assessment of system functions, (3) component safety-significance assessment, (4) assessment of DID and safety margin, (5) review by the IDP and final determination of safety-significance for system functions and components, (6) risk sensitivity studies to confirm that the risk acceptance guidelines of RG 1.174 are met, (7) periodic review to ensure continued categorization validity and acceptable performance for SSCs that have been categorized, and (8) documentation requirements identified in LAR Section 3.1.1. Procedures are formal plant documents, and changes will be tracked providing change control and records of the changes. These categorization documents and records, as described by the licensee, include documentation and record change controls consistent with NEI 00-04, endorsed by RG 1.201, and are in conformance with the requirements of 10 CFR 50.69(f)(1). Therefore, the NRC staff finds the documentation and records acceptable.

Based on its evaluation, the NRC staff finds that the change control and performance monitoring of categorized SSCs and PRA updates, as described in the LAR, as supplemented, will sufficiently capture and evaluate component failures to identify significant changes in the failure probabilities. In addition, the PRA update program and associated re-evaluation 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 that the licensee's adherence to the process in NEI 00-04 Sections 11 and 12 as described in 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 <u>Technical Conclusion</u>

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 and fire PRAs that, with the completion of the implementation items (summarized in Section 3.5.5 of this SE), will be of sufficient quality and level of detail to support the categorization process and that have been subjected to a peer review process against RG 1.200, Revision 2, as reviewed in Section 3.5.1 of this SE and, therefore, meets the requirements in 10 CFR 50.69(c)(1)(i);

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

(3) maintains defense-in-depth, 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 Section 3.8 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).

3.12 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 its RAIs, the NRC staff identified certain specific actions, as described below, that are necessary to support the staff's conclusion that the proposed program meets the requirements in 10 CFR 50.69 and the guidance in RG 1.201 and NEI 00-04.

The NRC staff's finding on the acceptability of the PRA evaluation in the licensee's proposed 10 CFR 50.69 process is conditioned on the completion of ten implementation items, nine of which are regarding updates to the internal events and fire PRA, and one of which regards the update of the SMA SSEL for the as-built, as operated plant. All of these items are identified in Attachment 1 of the licensee's letter dated September 28, 2018 (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 and PRA methods 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 1 to its September 28, 2018 letter (Reference 4), the licensee proposed the following condition to its license:

1. NextEra Energy Point Beach is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 structures, systems, and components (SSCs) using: Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events, including internal flooding, and internal fire; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 SSCs and their associated supports; and the results of non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards, i.e., seismic margin analysis (SMA) to evaluate seismic risk, and a screening of other external hazards updated using the external hazard screening significance process identified in ASME/ ANS PRA Standard RA-Sa-2009; as specified in License Amendment No. [NUMBER] dated [DATE].

2. Prior to implementation of the provisions of 10 CFR 50.69, NextEra Energy Point Beach shall complete the items below:

    a. Item A in Attachment 1, List of Categorization Prerequisites, to NextEra Energy Point Beach letter NRC 2017-0043, "License Amendment Request 287, Application to Adopt 10 CFR 50.69, 'Risk-Informed Categorization and Treatment of Structures, Systems, and Components (SSCs) for Nuclear Power Plants,'" dated August 31, 2017; and

b. Attachment 1, Point Beach 10 CFR 50.69 PRA Implementation Items, in NextEra Energy Point Beach letter NRC-2018-0044, "Supplement to Response to Request for Additional Information Regarding License Amendment Request 287, Application to Adopt 10 CFR 50.69, 'Risk informed Categorization and Treatment of Structures, System, and Components (SSCs) for Nuclear Power Plants,'" dated September 28, 2018.

3. 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. 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 is required by the proposed license conditions.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Wisconsin State official was notified of the proposed issuance of the amendments on November 19, 2018. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

These amendments change requirements with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20, or change surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously published a proposed finding that these amendments involve no significant hazards consideration and there has been no public comment on such finding (83 FR 6226). Accordingly, these 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 these amendments.

6.0 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.

7.0 REFERENCES

1. Coffey, R., NextEra Energy Company, letter to U. S. Nuclear Regulatory Commission, Point Beach Nuclear Plant, Units 1 and 2, Renewed Facility Operating License Nos. DPR-24 and DPR-27, NRC Docket Nos. 50-266 and 50-301, "Application to adopt 10 CFR 50.69, 'Risk-informed Categorization and Treatment of Structures, Systems, and Components (SSCs) for Nuclear Power Plants,'" August 31, 2017 (ADAMS Accession No. ML17243A201).

2. Coffey, R., NextEra Energy Company, letter to U. S. Nuclear Regulatory Commission, Point Beach Nuclear Plant, Units 1 and 2, Renewed Facility Operating License Nos. DPR-24 and DPR-27, NRC Docket Nos. 50-266 and 50-301, "Supplement to Application to Adopt 10 CFR 50.69, 'Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Plants," October 26, 2017 (ADAMS Accession No. ML17299A012).

3. Coffey, R., NextEra Energy Company, letter to U. S. Nuclear Regulatory Commission, Point Beach Nuclear Plant, Units 1 and 2, Renewed Facility Operating License Nos. DPR-24 and DPR-27, NRC Docket Nos. 50-266 and 50-301, "Response to Request for Additional Information Regarding License Amendment Request 287, Application to Adopt 10 CFR 50.69, 'Risk informed Categorization and Treatment of Structures, System, and Components (SSCs) for Nuclear Power Plants,'" August 10, 2018 (ADAMS Accession No. ML18222A539).

4. Craven, R., NextEra Energy Company, letter to U. S. Nuclear Regulatory Commission, Point Beach Nuclear Plant, Units 1 and 2, "Supplement to Response to Request for Additional Information Regarding License Amendment Request 287, Application to Adopt 10 CFR 50.69, 'Risk informed Categorization and Treatment of Structures, System, and Components (SSCs) for Nuclear Power Plants,'" September 28, 2018 (ADAMS Accession No. ML18271A114).

5. McCartney, Eric, NextEra Energy Company, letter to U.S. Nuclear Regulatory Commission, Point Beach Nuclear Plant, Units 1 and 2, Docket Nos. 50-266 and 50-301, Renewed License Nos. DPR-24 and DPR-27, "License Amendment Request 273, Application for Technical Specification Change Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program," dated July 3, 2014 (ADAMS Accession No. ML14190A267).

6. Rosenberg, Stacey, U.S. Nuclear Regulatory Commission, internal memorandum to Joseph Giitter, "Nuclear Regulatory Commission Report on Observations of Implementation of the

Industry Independent Assessment Team Close-out of Facts and Observations after Appendix X Issued," dated December 1, 2017 (ADAMS Accession No. ML17356A055).

7. Chawla, Mahesh, U.S. Nuclear Regulatory Commission, email to Steve Catron, NextEra Point Beach, LLC, "Final – Point beach – 50.69 Risk Informed LAR 287,Request for Additional Information, CAC Nos. MG0196 and MG0197, EPID-2017-LLA-0284," dated June 27, 2018 (ADAMS Accession No. ML18178A599).

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

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

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

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

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

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

14. Meyer, Larry, NextEra Energy Company, letter to U.S. Nuclear Regulatory Commission, Point Beach Nuclear Plant, Units 1 and 2, Docket Nos. 50-266 and 50-301, Renewed License Nos. DPR-24 and DPR-27, "License Amendment Request 271, Transition to 10 CFR 50.48(c) – NFPA 805, 'Performance Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants,' 2001 Edition," dated June 26, 2013 (ADAMS Accession No. ML13182A353).

15. McCartney, Eric, NextEra Energy Company, letter to U.S. Nuclear Regulatory Commission, Point Beach Nuclear Plant, Units 1 and 2, Docket Nos. 50-266 and 50-301, Renewed License Nos. DPR-24 and DPR-27, "NextEra Energy Point Beach, LLC, Response to NRC 10 CFR 50.54(f) Request for Information Regarding Near-term Task Force Recommendation 2.1, Flooding – Submittal of Flooding Hazards Revaluation Report," dated March 12, 2015 (ADAMS Accession No. ML15071A413).

16. Coffey, R., NextEra Energy Company, letter to U.S. Nuclear Regulatory Commission, Point Beach Nuclear Plant, Units 1 and 2, Docket Nos. 50-266 and 50-301, Renewed License Nos. DPR-24 and DPR-27, "NextEra Energy Point Beach, LLC, Focused Evaluation for Local Intense Precipitation," dated June 22, 2017 (ADAMS Accession No. ML17173A082).

17. Andersen, Victoria, Nuclear Energy Institute, letter to Stacey Rosenberg, 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).

18. Nuclear Energy Institute, "Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard," NEI 05-04, Revision 3, dated November 2009.

19. Nuclear Energy Institute, "Fire Probabilistic Risk Assessment (FPRA) Peer Review Process Guidelines," NEI 07-12, Revision 1, dated June 2010 (ADAMS Package Accession No. ML102230049).

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

21. Giitter, Joseph, and Ross-Lee, Mary Jane, U.S. Nuclear Regulatory Commission, letter to Greg Krueger, 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).

22. Pressurized Water Reactor Owners Group, "PRA Model for the Generation III Westinghosue Shutdown Seal," July 3, 2014 (ADAMS Accession No ML14190A331).

23. Morey, Dennis C., U.S. Nuclear Regulatory Commission, letter to W. Anthony Nowinowski, PWR Owners Group, "Final Safety Evaluation for Pressurized Water Reactor Owners Group Topical report PWROG-14001-P, Revision 1, 'PRA Model for the Generation III Westinghouse Shutdown Seal' (CAC No. MF4397)," dated August 23, 2017 (ADAMS Accession Nos. ML17200A116 and ML17200C876).

24. U.S. Nuclear Regulatory Commission, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making," NUREG-1855, Volume 1, dated March 2009 (ADAMS Accession No. ML090970525).

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

26. U.S. Nuclear Regulatory Commission, "Fire PRA Methodology for Nuclear Power Facilities," NUREG/CR-6850, Volumes 1 and 2, dated September 2005, and Supplement 1 to NUREG/CR-6850, "Fire Probatilistic Risk Assessment Methods Enhancements," dated September 2010 (ADAMS Accession Nos. ML052580075, ML052580118, and ML103090242, respectively).

27. Electric Power Research Institute, "A Methodology for Assessment of Nuclear Power Plant Seismic Margin," EPRI NP-6041-SL, Revision 1, dated August 1991.

28. U.S. Nuclear Regulatory Commission, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10 CFR 50.54(f)," Generic Letter 88-20, Supplement 4, dated June 28, 1991 (ADAMS Accession No. ML003769582 (non-public)).

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

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