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

February 28, 2020

Mr. Bryan C. Hanson
Senior Vice President
Exelon Generation Company, LLC
President and Chief Nuclear Officer
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: CALVERT CLIFFS NUCLEAR POWER PLANT, UNITS 1 AND 2 – ISSUANCE OF AMENDMENT NOS. 332 AND 310 RE: RISK-INFORMED CATEGORIZATION AND TREATMENT OF STRUCTURES, SYSTEMS, AND COMPONENTS FOR NUCLEAR POWER REACTORS (EPID L-2018-LLA-0482)

Dear Mr. Hanson:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 332 to Renewed Facility Operating License No. DPR-53 and Amendment No. 310 to Renewed Facility Operating License No. DPR-69 for the Calvert Cliffs Nuclear Power Plant, Units 1 and 2. These amendments consist of changes to the renewed facility operating licenses in response to your application dated November 28, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18333A022), as supplemented by letters dated November 29, 2018, and May 10, July 1, July 19, August 5, October 9, and October 30, 2019 (ADAMS Accession Nos. ML18337A038, ML19130A180, ML19183A012, ML19200A216, ML19217A143, ML19282B718, and ML19303A005, respectively).

These amendments allow the implementation of risk-informed process for the categorization and treatment of structures, systems, and components at Calvert Cliffs Nuclear Power Plant, Units 1 and 2.

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

Sincerely,

Michael L. Marshall, Jr., Senior Project Manager
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-317 and 50-318

Enclosures:
1. Amendment No. 332 to DPR-53
2. Amendment No. 310 to DPR-69
3. Safety Evaluation

cc: Listserv

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

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-317

CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 332
Renewed License No. DPR-53

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

   A. The application for amendment by Exelon Generation Company, LLC (Exelon, the licensee) dated November 28, 2018, as supplemented by letters dated November 29, 2018, and May 10, July 1, July 19, August 5, October 9, and October 30, 2019, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, by Amendment No. 332, Renewed Facility Operating License No. DPR-53 is hereby amended to authorize use of a risk-informed process for the categorization and treatment of structures, systems, and components as set forth in the licensee's application dated November 28, 2018, as supplemented by letters dated November 29, 2018, and May 10, July 1, July 19, August 5, October 9, and October 30, 2019, and evaluated in the NRC staff's safety evaluation enclosed with this amendment.

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

FOR THE NUCLEAR REGULATORY COMMISSION

James G. Danna, Chief
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility
Operating License

Date of Issuance: February 28, 2020

ATTACHMENT TO LICENSE AMENDMENT NO. 332

CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT 1

RENEWED FACILITY OPERATING LICENSE NO. DPR-53

DOCKET NO. 50-317

Replace the following pages of the renewed facility operating license with the attached revised pages. The revised pages are identified by amendment number and contain a marginal line indicating the areas of change.

Remove Page	Insert Page
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(4) Exelon Generation pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use, in amounts as required, any byproduct, source, and special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

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

C. This license is deemed to contain and is subject to the conditions set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act, and the rules, regulations, and orders of the Commission, now or hereafter applicable; and is subject to the additional conditions specified and incorporated below:

(1) Maximum Power Level

Exelon Generation is authorized to operate the facility at steady-state reactor core power levels not in excess of 2737 megawatts-thermal in accordance with the conditions specified herein.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 332, are hereby incorporated into this license. Exelon Generation shall operate the facility in accordance with the Technical Specifications.

(a) For Surveillance Requirements (SRs) that are new, in Amendment 227 to Facility Operating License No. DPR-53, the first performance is due at the end of the first surveillance interval that begins at implementation of Amendment 227. For SRs that existed prior to Amendment 227, including SRs with modified acceptance criteria and SRs whose frequency of performance is being extended, the first performance is due at the end of the first surveillance interval that begins on the date the Surveillance was last performed prior to implementation of Amendment 227.

(3) Additional Conditions

The Additional Conditions contained in Appendix C as revised through Amendment No. 327 are hereby incorporated into this license. Exelon Generation shall operate the facility in accordance with the Additional Conditions.

(4) Secondary Water Chemistry Monitoring Program

Exelon Generation shall implement a secondary water chemistry monitoring program to inhibit steam generator tube degradation. This program shall include:

a. Identification of a sampling schedule for the critical parameters and control points for these parameters;

b. Identification of the procedures used to quantify parameters that are critical to control points;

c. Identification of process sampling points;

d. Procedure for recording and management of data;

e. Procedures defining corrective actions for off control point chemistry conditions; and

f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events required to initiate corrective action.

(5) Mitigation Strategy

Exelon Generation shall develop and maintain strategies for addressing large fires and explosions that include the following key areas:

(a) Fire fighting response strategy with the following elements:

1. Pre-defined coordinated fire response strategy and guidance
2. Assessment of mutual aid fire fighting assets
3. Designated staging areas for equipment and materials
4. Command and control
5. Training of response personnel

(b) Operations to mitigate fuel damage considering the following:

1. Protection and use of personnel assets
2. Communications
3. Minimizing fire spread
4. Procedures for implementing integrated fire response strategy
5. Identification of readily available pre-staged equipment
6. Training on integrated fire response strategy
7. Spent fuel pool mitigation measures

(c) Actions to minimize release to include consideration of:

1. Water spray scrubbing
2. Dose to onsite responders

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

Exelon is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)-1, RISC-2, RISC-3,

and RISC-4 Structures, Systems, and Components (SSCs) using: Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events, including internal flooding, and internal fire; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 and non-Class SSCs and their associated supports; the results of the non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009 for other external hazards except seismic; and the alternative seismic approach as described in Exelon's original submittal letter dated November 28, 2018, and all its subsequent associated supplements as specified in License Amendment No. 332 dated February 28, 2020.

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

D. Exelon Generation 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: "Calvert Cliffs Nuclear Power Plant Security Plan, Training and Qualification Plan, and Safeguards Contingency Plan, Revision 1" submitted May 19, 2006.

Exelon Generation shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The licensee's CSP was approved by License Amendment No. 298 and modified by License Amendment No. 312.

E. Exelon Generation 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 September 24, 2013; as supplemented by letters dated February 9, 2015, March 11, 2015, April 13, 2015, July 6, 2015, August 13, 2015, February 24, 2016, and April 22, 2016, and as approved in the NRC safety evaluation dated August 30, 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), 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, 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 $1x10^{-7}$/yr for CDF and less than $1x10^{-8}$/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. 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 dated August 30, 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.

F. At the time of the next scheduled update to the FSAR required pursuant to 10 CFR 50.71(e)(4) following the issuance of this renewed license, Exelon Generation shall update the FSAR to include the FSAR supplement submitted pursuant to 10 CFR 54.21(d), as amended and supplemented by the program descriptions in Appendix E to the Safety Evaluation Report, NUREG-1705. Until that FSAR update is complete, Exelon Generation may make changes to the programs described in Appendix E without prior Commission approval, provided that the licensee evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

G. Any future actions listed in Appendix E to the Safety Evaluation Report, NUREG-1705, shall be included in the FSAR. Exelon Generation shall complete these actions by July 31, 2014, except for the volumetric inspections of the control element drive mechanisms, which must be completed no later than 2029 for Unit 1 (Appendix E, Item 65).

H. This renewed license is effective as of the date of issuance and shall expire at midnight on July 31, 2034.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Samuel J. Collins, Director
Office of Nuclear Reactor Regulation

Attachments:

Appendix A – Technical Specifications
Appendix B – Environmental Protection Plan (non-radiological) Technical Specifications
Appendix C – Additional Conditions

Date of Issuance: March 23, 2000

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

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-318

CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 310
Renewed License No. DPR-69

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

   A. The application for amendment by Exelon Generation Company, LLC (Exelon, the licensee) dated November 28, 2018, as supplemented by letters dated November 29, 2018, and May 10, July 1, July 19, August 5, October 9, and October 30, 2019, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, by Amendment No. 310, Renewed Facility Operating License No. DPR-69 is hereby amended to authorize use of a risk-informed process for the categorization and treatment of structures, systems, and components as set forth in the licensee's application dated November 28, 2018, as supplemented by letters dated November 29, 2018, and May 10, July 1, July 19, August 5, October 9, and October 30, 2019, and evaluated in the NRC staff's safety evaluation enclosed with this amendment.

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

FOR THE NUCLEAR REGULATORY COMMISSION

James G. Danna, Chief
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility
Operating License

Date of Issuance: February 28, 2020

ATTACHMENT TO LICENSE AMENDMENT NO. 310

CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT 2

RENEWED FACILITY OPERATING LICENSE NO. DPR-69

DOCKET NO. 50-318

Replace the following pages of the renewed facility operating license with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove Page	Insert Page
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(4) Exelon Generation pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use, in amounts as required, any byproduct, source, and special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

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

C. This license is deemed to contain and is subject to the conditions set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act, and the rules, regulations, and orders of the Commission, now or hereafter applicable; and is subject to the additional conditions specified and incorporated below:

(1) Maximum Power Level

Exelon Generation is authorized to operate the facility at steady-state reactor core power levels not in excess of 2737 megawatts-thermal in accordance with the conditions specified herein.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 310, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications.

(a) For Surveillance Requirements (SRs) that are new, in Amendment 201 to Facility Operating License No. DPR-69, the first performance is due at the end of the first surveillance interval that begins at implementation of Amendment 201. For SRs that existed prior to Amendment 201, including SRs with modified acceptance criteria and SRs whose frequency of performance is being extended, the first performance is due at the end of the first surveillance interval that begins on the date the Surveillance was last performed prior to implementation of Amendment 201.

(3) Less Than Four Pump Operation

The licensee shall not operate the reactor at power levels in excess of five (5) percent of rated thermal power with less than four (4) reactor coolant pumps in operation. This condition shall remain in effect until the licensee has submitted safety analyses for less than four pump operation, and approval for such operation has been granted by the Commission by amendment of this license.

(4) Environmental Monitoring Program

If harmful effects or evidence of irreversible damage are detected by the biological monitoring program, hydrological monitoring program, and the

1. Protection and use of personnel assets
2. Communications
3. Minimizing fire spread
4. Procedures for implementing integrated fire response strategy
5. Identification of readily available pre-staged equipment
6. Training on integrated fire response strategy
7. Spent fuel pool mitigation measures

(c) Actions to minimize release to include consideration of:

1. Water spray scrubbing
2. Dose to onsite responders

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

Exelon is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 Structures, Systems, and Components (SSCs) using: Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events, including internal flooding, and internal fire; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 and non-Class SSCs and their associated supports; the results of the non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009 for other external hazards except seismic; and the alternative seismic approach as described in Exelon's original submittal letter dated November 28, 2018, and all its subsequent associated supplements as specified in License Amendment No. 310 dated February 28, 2020.

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

D. Exelon Generation 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 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: "Calvert Cliffs Nuclear Power Plant Security Plan, Training and Qualification Plan, and Safeguards Contingency Plan, Revision 1" submitted dated May 19, 2006.

Exelon Generation shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The licensee's

CSP was approved by License Amendment No. 275 and modified by License Amendment No. 290.

E. Exelon Generation 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 September 24, 2013; as supplemented by letters dated February 9, 2015, March 11, 2015, April 13, 2015, July 6, 2015, August 13, 2015, February 24, 2016, and April 22, 2016, and as approved in the NRC safety evaluation dated August 30, 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), 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, 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 $1x10^{-7}$/yr for CDF and less than $1x10^{-8}$/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. 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 dated August 30, 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.

F. At the time of the next scheduled update to the FSAR required pursuant to 10 CFR 50.71(e)(4) following the issuance of this renewed license, Exelon Generation shall update the FSAR to include the FSAR supplement submitted pursuant to 10 CFR 54.21(d), as amended and supplemented by the program descriptions in Appendix E to the Safety Evaluation Report, NUREG-1705. Until that FSAR update is complete, Exelon Generation may make changes to the programs described in Appendix E without prior Commission approval, provided

that the licensee evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

G. Any future actions listed in Appendix E to the Safety Evaluation Report, NUREG-1705, shall be included in the FSAR. Exelon Generation shall complete these actions by August 13, 2016.

H. This renewed license is effective as of the date of issuance and shall expire at midnight on August 13, 2036.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Samuel J. Collins, Director
Office of Nuclear Reactor Regulation

Attachments:

Appendix A – Technical Specifications
Appendix B – Environmental Protection Plan (non-radiological) Technical Specifications
Appendix C – Additional Conditions

Date of Issuance: March 23, 2000

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 332 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-53

AMENDMENT NO. 310 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-69

EXELON GENERATION COMPANY, LLC

CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NOS. 1 AND 2

DOCKET NOS. 50-317 AND 50-318

1.0 INTRODUCTION

By application dated November 28, 2018 (Reference 1), as supplemented by letters dated November 29, 2018, and May 10, July 1, July 19, August 5, October 9, and October 30, 2019 (Reference 2, Reference 3, Reference 4, Reference 5, Reference 6, Reference 7, and Reference 8, respectively), Exelon Generation Company, LLC (Exelon or the licensee) submitted a license amendment request (LAR or the application) for the use of a risk-informed process for the categorization and treatment of structures, systems, and components at Calvert Cliffs Nuclear Power Plant, Units 1 and 2 (Calvert Cliffs). 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 U.S. Nuclear Regulatory Commission (NRC or the Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on January 30, 2019 (84 FR 494).

The amendments would modify the licensing basis to allow for the implementation of the provisions of Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.69, "Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors." The provisions of 10 CFR 50.69 allow adjustment of the scope of structures, systems, and components (SSCs) subject to special treatment requirements. Special treatment refers to those requirements that provide increased assurance beyond normal industry practices that SSCs will perform their design-basis functions. 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), the requirements may not be changed.

The risk-informed approach to regulation enhances and extends the traditional deterministic regulations by considering risk in a comprehensive manner. Specifically, a risk-informed 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. The use of PRA is one approach for categorizing SSCs.

2.0 REGULATORY EVALUATION

2.1 Proposed Changes

2.1.1 Proposed Risk-Informed Categorization Process

The licensee proposed a risk-informed categorization process for the implementation of the provisions of 10 CFR 50.69 that allow adjustment of the scope of SSCs subject to special treatment controls. If SSCs are determined to be of low safety significance using the proposed risk-informed categorization process, alternative treatment requirements can be implemented in accordance with this regulation. If SSCs are determined to be of high safety significance, treatment requirements will not be changed or enhanced. According to the licensee, this allows improved focus on equipment that has safety significance resulting in improved plant safety. With one exception, the licensee stated that the proposed a risk-informed categorization process is consistent with Nuclear Energy Institute (NEI) 00-04, Revision 0, "10 CFR 50.69 SSC Categorization Guideline" (Reference 9), dated July 2005, which was endorsed by the NRC in Regulatory Guide (RG) 1.201, Revision 1, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance," dated May 2006 (Reference 10). The exception identified by the licensee is the use of an alternative seismic approach.

2.1.2 Alternative Seismic Approach

The licensee's proposed alternative seismic approach is discussed in Section 3.2.3, "Seismic Hazards," of the enclosure to the licensee's letter dated May 10, 2019, and Attachment 2 to the licensee's letter dated July 19, 2019. The licensee's alternative seismic approach does not include a quantified consideration of seismic risk or SSC safety significance but rather uses the 10 CFR 50.69 categorization process, including the full power internal events PRA (FPIE) and other risk evaluations, along with the defense-in-depth (DID) and qualitative assessment by the integrated decision-making panel (IDP), to adequately identify the safety-significant functions and SSCs. To capture the potential impact of seismic risk in the categorization process, the licensee's alternative seismic approach includes qualitative assessments of plant SSC-specific seismic insights and its presentation to the IDP for consideration in its decisionmaking.

The licensee's alternative seismic approach is based on two important bases – the expectation that the seismic risk is low based on the seismic hazard at the plant, and the conclusions from the case studies in Electric Power Research Institute (EPRI) Report 3002012988. Based on the low seismic hazard, the licensee stated that it expected the seismic core damage frequency (CDF) and large early release frequency (LERF) estimates for its site to be low and to not lead to SSCs being categorized as high safety-significant solely from seismic risk contribution.

The licensee compared the reevaluated seismic hazard for Calvert Cliffs developed in response to Near Term Task Force (NTTF) Recommendation 2.1 (Reference 31) against the site's seismic design-basis safe shutdown earthquake (SSE) to demonstrate that the following criteria is met:

> GMRS [Ground Motion Response Spectrum] peak acceleration is at or below approximately 0.2g or where the GMRS is below or approximately equal to the SSE between 1.0 Hz [hertz] and 10 Hz.

Based on the above-mentioned criteria being met for Calvert Cliffs, the licensee stated that it demonstrated the low seismic hazard at its site, and therefore, justified the use of the alternative seismic approach.

In the November 28, 2018, application, the licensee stated that EPRI 3002012988 applied to Calvert Cliffs in its entirety except for Sections 2.3 and 2.4 of Appendices A and B. The licensee's alternative seismic approach relied on conclusions from case studies performed in EPRI Report 3002012988. The case studies compared HSS SSCs identified from seismic PRAs (SPRAs) against HSS SSCs identified from FPIEs and, in certain cases, fire PRAs (FPRAs) for four different plants. The case studies were used to identify conclusions on the determination of safety significance of SSCs uniquely from SPRAs (i.e., the SSCs identified as HSS that were not identified as HSS in the FPIEs and, as applicable, FPRAs).

The NRC staff notes that Figure 2-2 of EPRI Report 3002012988 replicates Figure 2-1 and appears to be a typographical error. However, the error did not impact the NRC staff's review of the proposed alternative seismic approach.

2.1.3 Licensee Proposed License Condition

In addition, the licensee proposed to amend its renewed facility operating licenses by adding the following license condition:

> Exelon is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 Structures, Systems, and Components (SSCs) using: Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events, including internal flooding, and internal fire; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (AN0-2) passive categorization method to assess passive component risk for Class 2 and Class 3 and non-Class SSCs and their associated supports; the results of the non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009 for other external hazards except seismic; and the alternative seismic approach as described in Exelon's original submittal letter dated November 28, 2018, and all its subsequent associated supplements as specified in License Amendment No. [XXX] dated [DATE].
>
> 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).

2.2 Regulatory Requirements

Section 50.69 of 10 CFR provides an alternative approach for establishing requirements for treatment of SSCs for nuclear power reactors using an integrated and systematic risk-informed process for categorizing SSCs according to their safety significance. Specifically, for SSC

categorized as LLS, alternative treatment requirements may be implemented in accordance with the regulation. For SSCs determined to be HSS, requirements may not be changed.

Section 50.69(b) of 10 CFR, "Applicability and scope of risk-informed treatment of SSCs and submittal/approval process," specifies the requirements for information in an application for a license amendment to voluntarily adopt 10 CFR 50.69. Section 50.69(b)(2)(ii) of 10 CFR requires the following information:

> A description of the measures taken to assure that the quality and level of detail of the systematic processes that evaluate the plant for internal and external events during normal operation, low power, and shutdown (including the plant-specific probabilistic risk assessment (PRA), margins-type approaches, or other systematic evaluation techniques used to evaluate severe accident vulnerabilities) are adequate for the categorization of SSCs.

The statements of considerations (SoC) accompanying the publication of 10 CFR 50.69 states that if a licensee wishes to use an approach different from that in NEI 00-04, as endorsed by RG 1.201, Revision 1, the submittal must provide a sufficient description of how the categorization would be conducted. The SoC further states that, as part of the submittal, a licensee or applicant must also describe what measures it has used for the methods other than a PRA to determine its adequacy for this application.

The regulation in 10 CFR 50.69(b)(2)(iv) requires, in part, the following information:

> A description of, and basis for acceptability of, the evaluations to be conducted to satisfy § 50.69(c)(1)(iv).

Section 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 in Appendix J to 10 CFR Part 50, and (xi) certain requirements of Appendix A to 10 CFR Part 100.

Section 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).

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

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

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

(iii) Maintain defense-in-depth.

(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 CDF and LERF resulting from changes in treatment permitted by implementation of §§ 50.69(b)(1) and (d)(2) are small.

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

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[1] NEI 00-04 uses the term "high-safety-significant (HSS)" to refer to SSCs that perform safety-significant functions. The NRC staff 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.

The SoC on 10 CFR 50.69(c)(1)(iv) states that if a PRA model does not exist for the external initiating events or the low power and shutdown operating modes, justification should be provided, on the basis of bounding analyses or qualitative considerations, that the effect on risk (from the unmodeled events or modes of operation) is not significant and that the total effect on risk from modeled and unmodeled events and modes of operation is small, "consistent with Section 2.2.4 of RG 1.174."

Section 50.69(c)(2) of 10 CFR states:

> The SSCs must be categorized by an Integrated Decision-Making Panel (IDP) staffed with expert, plant-knowledgeable members whose expertise includes, at a minimum, PRA, safety analysis, plant operation, design engineering, and system engineering.

2.3 Regulatory Guidance

NEI 00-04, Revision 0, describes a process for determining the safety significance of SSCs and categorizing them into the four RISC categories defined in 10 CFR 50.69(a). This categorization process is an integrated decision-making process that incorporates risk and traditional engineering insights. NEI 00-04, Revision 0, provides options for licensees implementing different approaches depending on the scope of their PRA models. It also allows for the use of non-PRA approaches when PRA models have not been developed to address hazards such as seismic, fire, or shutdown risk.

The NEI 00-04 guidance identifies non-PRA methods to be used as an approach, such as fire-induced vulnerability evaluation to address internal fire risk, seismic margin analysis (SMA) to address seismic risk, and guidance in the Nuclear Management and Resources Council (NUMARC) report NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management," December 1991 (Reference 26), to address shutdown operations. As stated in RG 1.201, such non-PRA-type evaluations will result in more conservative categorization, in that special treatment requirements will not be allowed to be relaxed for SSCs that are relied upon in such evaluations. The degree of relief that the NRC will accept under 10 CFR 50.69 (i.e., SSCs subject to relaxation of special treatment requirements) will be commensurate with the assurance provided by the evaluations performed to assess and characterize the SSC's risk.

Sections 2 through 10 of NEI 00-04 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(f). Section 12 of NEI 00-04 provides guidance on periodic review related to the requirements of 10 CFR 50.69(e). Maintaining change control and periodic review provides confidence that all aspects of the program reasonably reflect the current as-built, as-operated plant configuration and operating practices, and applicable plant and industry operational experience, as required by 10 CFR 50.69(c)(1)(ii).

Sections 1.5 and 5.3 of NEI 00-04 identify either a plant-specific SPRA or a SMA that reflects the current as-built, as-operated plant as acceptable approaches for identifying SSCs that are safety-significant due to seismic risk in the categorization process. In Section 3.2.5 of the enclosure to the LAR, as well as Attachment 1 of the May 10, 2019, supplement, the licensee proposed the use of an alternative approach for considering the seismic risk in the categorization process. The licensee stated that EPRI Report 3002012988, "Alternative Approaches for Addressing Seismic Risk in 10 CFR 50.69 Risk-Informed Categorization" (Reference 11), provides the technical basis for its proposed alternative seismic approach. The proposed alternative seismic approach constitutes a deviation from the guidance endorsed in RG 1.201, Revision 1.

RG 1.201, Revision 1, endorses the categorization process described in NEI 00-04, Revision 0, with clarifications, limitations, and conditions. RG 1.201, Revision 1, states that the applicant is expected to document, at a minimum, the technical adequacy of the internal initiating events PRA. Licensees may use either PRAs or alternative approaches for hazards other than internal initiating events. The guidance in RG 1.201, Revision 1, clarifies that the NRC staff expects that licensees proposing to use non-PRA approaches in their categorization should provide a basis in the submittal explaining why the approach and the accompanying method employed to assign safety significance to SSCs is technically acceptable.

RG 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities" (Reference 12), describes an acceptable approach for determining whether the acceptability of the PRA, in total, or the parts that are used to support an application, is sufficient to provide confidence in the results such that the PRA can be used in regulatory decisionmaking for light-water reactors. It endorses, with clarifications, the 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 13). 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 14), 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.

NUREG-1855, Revision 1, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking," dated March 2017 (Reference 15), provides guidance on how to treat uncertainties associated with PRA in risk-informed decisionmaking. The guidance fosters an understanding of the uncertainties associated with PRA and their impact on the

results of the PRA and provides a pragmatic approach to addressing these uncertainties in the context of the decisionmaking.

3.0 TECHNICAL EVALUATION

3.1 NRC Staff's Method of Review

The NRC staff reviewed the licensee's application to determine if the proposed changes are consistent with the regulations and guidance discussed in Section 2.0 of this safety evaluation (SE). The NRC staff's review and the documentation of that review in this SE uses the framework of NEI 00-04, Revision 0, as endorsed in RG 1.201, Revision 1.

Regulatory guidance is not a substitute for regulations, and compliance with them is not required. For methods and solutions that differ from those set forth in regulatory guidance, the NRC staff's review focuses on the determination of a sufficient basis to make the applicable regulatory findings. The NRC staff's review of the licensee's proposed alternative seismic approach focused on (1) whether it met the requirements for categorization approaches set forth in 10 CFR 50.69(b) and (c), as clarified by the SoC for 10 CFR 50.69 (69 FR 68047); (2) the technical acceptability of the PRAs (i.e., SPRA, as well as FPIE and, as applicable, FPRA) used for the case studies in EPRI Report 3002012988 that supports the proposed alternative seismic approach; (3) the acceptability of the conclusions from the case studies in EPRI Report 3002012988; (4) the acceptability of the implementation of the conclusions from the case studies by the licensee; and (5) the performance monitoring supporting the proposed alternative seismic approach to meet the requirements of 10 CFR 50.69(e).

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

The guidance in RG 1.201, Revision 1, 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. Defense-in-Depth Assessment (Section 6 of NEI 00-04)
5. Preliminary Engineering Categorization of Functions (Section 7 of NEI 00-04)
6. Risk Sensitivity Study (Section 8 of NEI 00-04)
7. IDP Review and Approval (Section 9 of NEI 00-04)
8. SSC Categorization (Section 10 of NEI 00-04)

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

As generally set forth in the LAR, the licensee's SSC risk-informed categorization process contains the following elements:

- Defining system boundaries.
- Defining system functions and assigning components to functions.
- Risk Characterization. Safety significance of active components is assessed through a combination of PRA and non-PRA methods, covering all hazards.
- DID characterization performed in accordance with Section 6 of NEI 00-04.
- 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.
- Qualitative Characterization. System functions are qualitatively categorized as HSS or LSS based on the seven questions in Section 9.2 of NEI 00-04.
- 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.
- 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.

In the LAR, the licensee explained that consistent with NEI 00-04, the categorization of a component or function is "preliminary" until it has been confirmed by the IDP. The licensee stated that a component or function is preliminarily categorized as HSS, if any element of the process results in a preliminary HSS determination. This preliminary categorization will be presented to the IDP for review. The IDP will decide the final categorization.

In Table 3-1 of the LAR, the licensee described 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).

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), the DID assessment (Section 6 of NEI 00-04), or the qualitative criteria (Section 9 of NEI 00-04), the associated system function(s) would be identified as HSS. Once a system function is identified as HSS, then all the components supporting that function are deemed preliminary HSS and will be presented to the IDP for review.

The NRC staff has evaluated the categorization steps 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, Revision 1.

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 it is adequate to support this application. The guidance in Section 3 of NEI 00-04 summarizes the use of risk information and the general quality measures that should be applied to the risk analyses supporting the 10 CFR 50.69 categorization, 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 the EPRI alternative approach described in EPRI Report 3002012988 to assess seismic risk, its individual plant examination of external events (IPEEE) screening to assess the risk from other external hazards (high winds, external floods), and its shutdown safety plan to assess shutdown risk.

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

Section 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.

Section 4 of NEI 00-04 also states that a candidate LSS SSC that supports an interfacing system "will remain uncategorized until the interfacing system is considered." In its letter dated October 9, 2019, the licensee proposed to categorize an SSC that supports functions in an interfacing system without completing the categorization of that interfacing system, if the following two conditions are met: (1) an interface SSC failure cannot prevent performance of interface system functions, and (2) the risk is limited to passive failures assessed as LSS, following the passive categorization process for the applicable pressure boundary segments. The licensee stated that the interface SSC:

> [...] can be assessed without performing a full interface system categorization because adequate interface system function knowledge is available to perform the functional assessment and passive risk assessment. Categorizing the entire interfacing system would produce the same functional assessment and passive risk significance for the component.

The NRC staff notes that the passive failure classification proposed by the licensee only affects treatment programs for Class 2 and Class 3 pressure-retaining items and their associated supports (exclusive of Class CC and MC items). Passive failures are not normally modeled in PRAs, and the licensee's proposed passive categorization process relies on the conditional core damage and large early release probabilities following a passive failure, which are determined by imposing the impact of the passive failure on all components modeled in the PRA. This passive categorization method requires the full impact of the passive failure on safety significance to be evaluated, regardless of which system the component is assigned to. In addition to the passive categorization method, the licensee also stated it will perform the categorization only if it can confirm that a failure of the interface component cannot prevent performance of an interfacing system function. The NRC staff finds that the licensee's proposal, as described above, will yield the same or more conservative results when (and if) the uncategorized system is categorized, and therefore, the NRC staff accepts the licensee's proposal.

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

Section 2.2 of the LAR states that the safety functions in the categorization process include the design-basis functions, as well as functions credited for severe accidents (including external events). Section 3.1.1 of the LAR summarizes the different hazards and plant states for which functional and risk-significant information will be collected. In addition, Section 3.1.1 of the LAR states that the SSC categorization process documentation will include, among other items, system functions identified and categorized with the associated bases and mapping of components to support function(s).

Section 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)

Section 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 the following hazard groups:

- Internal events (including internal flooding)
- Fire
- Seismic
- Other external hazards (e.g., tornadoes, external flooding)
- Shutdown events

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

In LAR Sections 3.1.1 and 3.2.1 through 3.2.5, the licensee stated that the categorization process uses PRA to assess risks for the 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:

- EPRI alternative approach to assess seismic risk
- IPEEE screening to assess the risk from other external hazards (high winds, external floods)
- Shutdown safety plan to assess low power and shutdown risk
- Passive components: Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization methodology

The approaches and methods used by the licensee to assess internal events, other external hazards, and shutdown events, are consistent with the methods included in the NEI 00-04 guidance, as endorsed by RG 1.201, Revision 1. The non-PRA method for the categorization of passive components is consistent with the ANO-2 methodology for passive components (Reference 16) approved for risk-informed safety classification and treatment for repair and replacement activities in Class 2 and Class 3 moderate and high energy systems. The NEI 00-04 guidance, as endorsed by RG 1.201, Revision 1, considers the results and insights from the plant-specific PRA peer reviews as required by 10 CFR 50.69(c)(1)(i), and non-PRA risk characterization as required by 10 CFR 50.69(c)(1)(ii). To address seismic events, the

licensee proposed to use an alternative method not specified in the NEI 00-04 guidance, as endorsed by RG 1.201, Revision 1.

3.5.1 Evaluation of PRA Acceptability to Support the SSC 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) an FPRA.
Section 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. Section 50.69(b)(2)(iii) of 10 CFR requires the results of the peer review process conducted to meet 10 CFR 50.69(c)(1)(i) be submitted as part of the application.

3.5.1.1 Internal Events PRA

The licensee stated in LAR Section 3.2 that the PRA models are the same PRA models credited in Exelon's Calvert Cliffs LAR to adopt Technical Specifications Task Force Traveler, TSTF-505, Revision 1, "Provide Risk-Informed Extended Completion Times - RITSTF [Risk-Informed TSTF] Initiative 4b," with routine maintenance updates applied. The NRC staff's review of that amendment request is documented in an SE dated October 30, 2018 (Reference 17). In its letter dated July 1, 2019, the licensee stated that FLEX (Diverse and Flexible Coping Strategy) equipment and associated operator actions are not currently credited in the internal events or the FPRA. Therefore, the NRC staff's review of the internal events and flooding PRAs was based on the results provided in the LAR and TSTF-505 application.

As stated in the LAR and TSTF-505 application, a full scope peer review was performed in June 2010 for the internal events PRA (including internal flooding) against the requirements of the ASME/ANS 2009 Standard and RG 1.200, Revision 2. A focused-scope peer review was conducted in January 2017 for a PRA upgrade related to changes in the internal flooding PRA, including changes to the pipe break rupture frequencies, and was performed against the ASME/ANS 2009 Standard using the process in NEI 05-04, Revision 3, "Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard," dated November 2009 (Reference 18). A finding closure review was conducted by the licensee on the internal events PRA model in January 2017. Closed findings were reviewed and closed using the process documented in Appendix X to NEI 05-04, NEI 07-12, and NEI 12-13 as accepted by NRC in letter dated May 3, 2017 (Reference 19).

In Attachment 3 of the LAR, the licensee provided and dispositioned two facts and observations (F&Os) related to internal flooding that remained open after the Appendix X F&O closure. The NRC staff finds the disposition of the two F&Os acceptable for this application since they are documentation related, and therefore, do not impact the staff's decision.

Based on the NRC staff's review and on the previous review of the licensee's application for TSTF-505 (Reference 17), the NRC staff finds that the internal events and internal flooding PRA have been adequately peer reviewed against the current version of the PRA standard and RG 1.200, and that the licensee has adequately dispositioned the F&Os to support the technical adequacy of the internal events PRA for the Calvert Cliffs 50.69 risk-informed categorization program.

3.5.1.2 FPRA

The licensee stated in the LAR that the PRA models credited in the 50.69 LAR are the same PRA models credited in the NRC staff's SE concerning the approval of the licensee's risk-informed, performance-based fire protection program at Calvert Cliffs (Reference 20) with routine maintenance updates applied. Therefore, the NRC staff's review of the FPRA was based on the results provided in the LAR and TSTF-505 application.

The licensee evaluated the technical adequacy of the Calvert Cliffs FPRA model by conducting a full-scope peer review in January 2012 using NEI 07-12, Revision 1, "Fire Probabilistic Risk Assessment (FPRA) Peer Review Process Guidelines, dated June 2010 (Reference 21), and the FPRA (Part 4) of the PRA standard, as clarified by RG 1.200, Revision 2.

As a result of its review of the National Fire Protection Association (NFPA) 805 LAR dated September 24, 2013 (Reference 22), as supplemented, the NRC staff concluded, in the issuance of Amendment Nos. 318 and 296 dated August 30, 2016 (Reference 20), that (1) the FPRA model adequately represents the current as-built, as-operated configuration, and is, therefore, capable of modeling the plant as needed; (2) the FPRA model conforms sufficiently to the applicable industry PRA standards at an appropriate capability category, considering the acceptable disposition of the peer review and NRC staff review findings; and (3) the fire modeling used to support the development of the FPRA has been confirmed as appropriate and acceptable. A similar conclusion was reached during the NRC staff's review of the TSTF-505 application. During its review of this LAR, the NRC staff identified no information that would invalidate the NRC staff's NFPA 805 or TSTF-505 conclusion that the FPRA is technically acceptable to support risk calculations. Therefore, the NRC staff concludes that the FPRA is technically acceptable to support the 50.69 program.

3.5.2 Importance Measures and Sensitivity Studies

Section 50.69(c)(1)(i) of 10 CFR requires the licensee to consider the results and insights from the PRA during its categorization. These requirements are met, in part, by using importance measures and sensitivity studies as described in the methodology in NEI 00-04, Section 5. Fussell-Vesely and Risk Achievement Worth importance measures are obtained for each component and each PRA-modeled hazard (i.e., separately for the internal events PRA and for the FPRA), 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 the integrated measures exceed the criteria are assigned preliminary HSS.

The guidance in NEI 00-04 specifies the sensitivity studies to be conducted for each PRA model. The sensitivity studies are performed to ensure that assumptions associated with these specific uncertain parameters (i.e., human error, common-cause failure, and maintenance probabilities) are not masking the importance of a component. The NEI 00-04 guidance states that any additional "applicable sensitivity studies" from characterization of PRA adequacy should be considered.

In LAR Section 3.2.7, the licensee stated that it used the detailed process of identifying, characterizing, and qualitatively screening of model uncertainties found in Section 5.3 of NUREG-1855, Revision 0 (Reference 23), and Section 3.1.1 of EPRI Technical Report

(TR)-1016737, "Treatment of Parameter and Modeling Uncertainty for Probabilistic Risk Assessments" (Reference 24).

In LAR Attachment 6, the licensee provided a list of key assumptions and sources of modeling uncertainties that were reviewed for the internal events (including internal flooding) and FPRAs 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. In its letter dated July 19, 2019, the licensee reviewed NUREG-1855, Revision 1 (Reference 15), to assess the process and criteria used to identify the application-specific key assumptions and sources of uncertainty. The review of NUREG-1855, Revision 1, identified the need to systematically assess the generic FPRA uncertainties in Appendix B of EPRI TR-1026511 (Reference 25). The licensee stated that this additional assessment of key sources of fire generic uncertainties was performed and no new potential key sources of model uncertainty or assumptions were identified because of the review. Additional steps of NUREG-1855, Revision 1, were reviewed and determined to either have been addressed by the licensee's previous methodology or did not apply, since the licensee employed a more conservative methodology.

Further, in its letter dated July 19, 2019, the licensee provided a description of the process and criteria used to identify the application-specific key assumptions and sources of uncertainties. The licensee stated that as part of the model uncertainty analyses, the potential key assumptions and uncertainties identified for each ASME PRA element are reviewed to identify those uncertainties and assumptions that "may have the potential to affect the results in order to determine that reasonable alternative assumptions, if available, do not affect the decision" and aggregated in the PRA Uncertainty Assessment Notebook. The licensee further stated that for the 50.69 analysis, the assumptions and uncertainties and their characterizations in the internal events and FPRA Uncertainty Assessment Notebooks were reviewed specific to the application. Human error was identified as the candidate key source of model uncertainty for the 50.69 application. This uncertainty is addressed in NEI 00-04 by requiring sensitivity studies that increase internal events human error basic events to the 95th percentile and also decrease them to the 5th percentile.

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 FPRAs consistent with the guidance in RG 1.200, Revision 2, NUREG-1855, and EPRI TR-1026511, as applicable. The NRC staff finds that the requirement specified in 10 CFR 50.69(c)(1)(i) concerning consideration of the results and insights from the PRA during categorization is met because the licensee's proposed process considers integrated importance measures, sensitivity studies, and uncertainty consistent with NEI 00-04, as endorsed by RG 1.201, Revision 1.

3.5.3 Non-PRA Methods

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

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

- EPRI alternative approach to assess seismic risk
- IPEEE screening to assess the risk from other external hazards (high winds, external floods)
- Shutdown Safety Plan as described in NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management" (Reference 26), to assess low power and shutdown risk.

3.5.3.1 Seismic Risk

The licensee provided a description of its proposed alternative seismic approach for considering seismic risk in the categorization process and how the proposed alternative seismic approach would be used in the categorization process in Section 3.2.3 of the enclosure to the LAR and Attachment 1 to the licensee's letter dated May 10, 2019. In addition, the licensee based the acceptability of its proposed alternative seismic approach on the conclusions gained from case studies performed in EPRI Report 3002012988, and therefore, indirectly, on the acceptability of the PRAs used for the case studies. The information presented in the LAR and supplements, as well as in EPRI Report 3002012988, taken together, provides sufficient detail for the proposed alternative seismic approach, how the licensee's proposed alternative seismic approach would be used in the categorization process, and the measures for assuring the quality and level of detail for the licensee's proposed alternative seismic approach are adequate for the categorization of SSCs. Therefore, the NRC staff finds that the requirements in 10 CFR 50.69 (b)(2)(ii) are met for the proposed alternative seismic approach.

EPRI Report 3002012988 includes the results from case studies performed to determine the extent and type of unique HSS SSCs from SPRAs. The case studies were performed for four plants designated Plants A through D in EPRI Report 3002012988. Information regarding the case study plants and the PRAs used for each plant as part of the corresponding case study is provided in Table 1 below. The case study plants will be referred to by their designators A through D.

Each case study consisted of a comparison of the HSS SSCs categorized using the SPRA for that plant against HSS SSCs categorized using the corresponding FPIE PRA and, as applicable, FPRA. The comparison used mapping to assign SSCs and SSC failure modes from the SPRA to the FPIE or FPRA. The purpose of the mapping was to identify SSCs and SSC failure modes from the SPRA that were not captured by the FPIE and, as applicable, FPRA.

Table 1: Summary of Case Study Plants in EPRI Report 3002012988

Case Study Plant	Nuclear Steam Supply System and Containment Type	PRAs Exercised for Case Study
A (Peach Bottom Atomic Power Station, Units 1 and 2)	BWR/4, Mark I	SPRA, FPIE, and FPRA
B (North Anna Power Station, Units 1 and 2)	Westinghouse 3-loop, Large Dry Subatmospheric	SPRA and FPIE
C (Vogtle Electric Generating Station, Units 1 and 2)	Westinghouse 4-loop, Large Dry Atmospheric	SPRA, FPIE, and FPRA
D (Watts Bar Nuclear Plant, Units 1 and 2)	Westinghouse 4-loop, Ice Condenser	SPRA and FPIE

Section 3.6 of EPRI Report 3002012988 provides a summary of the conclusions derived from the four case studies. The report concludes that the case studies revealed that the only SSCs identified as HSS in the SPRA that were not also HSS from FPIE or FPRA were from unique seismically-induced failure modes. In its letter dated July 19, 2019, Exelon provided additional information to further explain the conclusions derived from the four case studies.

Section 3.4.3 of Attachment 1 to the licensee's letter dated May 10, 2019, included additional information on how the proposed alternative seismic approach meets the requirements of Section 50.69(c)(1)(iv). The information presented in the LAR and supplements, taken together, provides a sufficient description of, and basis for acceptability of, the evaluations to be conducted to satisfy 10 CFR 50.69(c)(1)(iv) for the alternative seismic approach. Therefore, the NRC staff finds that the requirements in 10 CFR 50.69 (b)(2)(iv) are met for the proposed alternative seismic approach.

Evaluation of Technical Acceptability of the PRAs Used for Case Studies Supporting the Proposed Alternative Seismic Approach

In Attachment 1 of the May 10, 2019, supplement, the licensee incorporated by reference into its application information related to Plant A (Reference 27) and Plant D (Reference 28, Reference 29, and Reference 30). The NRC staff's review of the technical acceptability of the PRAs used for the case studies focused on determining whether major deficiencies existed in the PRAs used for the case studies, and consequently, the conclusions developed from those case studies. During its review, the NRC staff determined that the Plant B case study was not essential for making the NRC staff's regulatory and technical conclusions on the proposed alternative seismic approach. Therefore, the Plant B case study, as well as the PRAs used for that case study, were not reviewed by the NRC staff for this application.

Brief descriptions of the peer reviews performed for the PRAs supporting the Plant A, C, and D case studies were provided in EPRI Report 3002012988. Sections 3.2.2.1, 3.2.3.1, and 3.2.4 contain this information for Plant A. Similar information is discussed in Section 3.4.1 for Plant C. Information on the peer reviews for PRAs used in the case study for Plant D was not provided in EPRI Report 3002012988. A summary of the regulatory activities for which the SPRA, FPIE, and FPRA, as applicable, for each case study plant was credited is provided in Table 2 below.

Table 2: Summary of PRA Use for Regulatory Actions by Relevant Case Study Plants

Case Study Plant	PRA Exercised for Case Study	Regulatory Activity Supported by PRA
A (Peach Bottom Atomic Power Station, Units 1 and 2)	SPRA	March 2012 10 CFR 50.54(f) request arising from NTTF Recommendation 2.1 (Reference 31)
	FPIE	10 CFR 50.69 LAR
	FPRA	10 CFR 50.69 LAR
C (Vogtle Electric Generating Station, Units 1 and 2) C (Vogtle Electric Generating Station, Units 1 and 2)	SPRA	10 CFR 50.69 LAR; March 2012 10 CFR 50.54(f) request arising from NTTF Recommendation 2.1
	FPIE	Risk-Informed Completion Time (RICT) LAR; 10 CFR 50.69 LAR
	FPRA	RICT LAR; 10 CFR 50.69 LAR
D (Watts Bar Nuclear Plant, Units 1 and 2)	SPRA	10 CFR 50.69 LAR; TSTF-425 LAR; March 2012 10 CFR 50.54(f) request arising from NTTF Recommendation 2.1
	FPIE	10 CFR 50.69 LAR; TSTF-425 LAR

Plant A - Peer-Review Process and Resolution of Peer-Review Findings

Section 3.2.2.1 of EPRI Report 3002012988 discussed the peer review of the Plant A SPRA that was performed in March 2017. Additional details regarding the Plant A SPRA were available to the NRC staff from publicly available information for Plant A submitted as part of the response for Plant A to a 10 CFR 50.54(f) request issued by the NRC in March 2012 arising from NTTF Recommendation 2.1, which was incorporated by reference by Exelon for this application in its letter dated May 10, 2019. Appendix A of the Plant A response to the 10 CFR 50.54(f) request contains detailed information on the peer review for the Plant A SPRA, as well as the finding level F&Os and the corresponding dispositions.

The description and basis of the F&O 3-1 related to supporting requirements (SRs) SFR-A2 and SPR-C1 indicated that the approach taken at the time of the peer review to identify dominant contributors for possible improvements lacked realism. The licensee's disposition of the finding was unclear on whether the changes made to the Plant A SPRA were included in the SPRA used to develop categorization conclusions from the Plant A case study. In its letter dated July 19, 2019, the licensee included an explanation of multiple sensitivity studies performed to quantify the integrated risk impacts of improvements such as human reliability analysis refinement, credit for FLEX equipment and actions, and refinement of fragility calculations. In addition, the licensee confirmed that changes made to the Plant A SPRA to disposition F&O 3-1 were included in the SPRA used for the case study. A review of the integrated sensitivity studies was performed to determine the combinations of integrated changes in the SPRA.

The NRC staff reviewed the information provided by the licensee, as well as the discussion of the Plant A SPRA in response to the 10 CFR 50.54(f) letter. Based on its review, the NRC staff finds that a systematic approach was followed for identifying and addressing significant seismic failures for Plant A SPRA and that the approach considered the combined impact of the sets of failures. Therefore, the NRC staff concludes that F&O 3-1 is dispositioned for the Plant A case study supporting the proposed alternative seismic approach. As part of its review, the NRC staff

also found that the open finding level F&Os for the Plant A FPIE were either adequately dispositioned for the SPRA or did not impact the SPRA.

In its letter dated July 19, 2019, the licensee provided the results of a sensitivity case that refined the highest acceleration 'bin' for the SPRA (i.e., divided it into multiple smaller 'bins') and eliminated credit for FLEX equipment and actions for those 'bins.' The sensitivity case resulted in certain relay-chatter failures becoming high safety-significant. The NRC staff's review of the sensitivity case finds that new conclusions were not generated because relay chatter was identified as a unique seismically-induced failure mode in EPRI Report 3002012988 and included in the proposed alternative seismic approach.

The Plant A SPRA peer review was performed using the guidance in NEI 12-13 and the requirements in Part 5 of the 2013 ASME/ANS PRA Standard. RG 1.200, Revision 2, endorses the 2009 ASME/ANS PRA Standard and does not endorse the 2013 ASME/ANS PRA Standard. The NRC staff has previously reviewed the differences between the SRs in Part 5 of the 2013 ASME/ANS PRA Standard against those in Part 5 of the 2009 ASME/ANS PRA Standard and found the use of Part 5 of the 2013 ASME/ANS PRA Standard (also known as Addendum B) to be an acceptable alternative to the NRC-endorsed approach for the Plant C SPRA used to support categorization of SSCs under 10 CFR 50.69 (Reference 35). In addition, the NRC staff also reviewed SRs in Part 5 of the 2013 ASME/ANS PRA Standard that may have plant-specific differences using the information available for the Plant A SPRA. Therefore, the NRC staff included consideration of the differences between the SRs in Part 5 of the 2013 ASME/ANS PRA Standard and in the 2009 ASME/ANS PRA Standard in the review of the technical acceptability of the Plant A SPRA for the case study supporting the proposed alternative seismic approach. The purpose of determining the technical acceptability of Plant A SPRA for this application is to determine the appropriateness of the conclusions from the corresponding case study and not for a Plant A-specific licensing action. Based on the information available to the NRC staff, as well as its review, the staff finds that use of Part 5 of the 2013 ASME/ANS PRA Standard does not impact the technical acceptability of the Plant A SPRA for use in the case study.

Section 3.2.3.1 of EPRI Report 3002012988 stated that the Plant A FPIE was peer-reviewed in November 2010. Section 3.2.4.1 of EPRI Report 3002012988 stated that the Plant A FPRA was peer-reviewed in November 2011, and the F&O independent assessment team finding closure review was conducted in November 2016. The NRC staff previously reviewed the results of the Plant A FPIE and FPRA peer reviews, including the independent F&O closure review for the FPRA, as part of the Plant A 10 CFR 50.69 LAR (Reference 32). The review included an evaluation of the dispositions of the peer review results and concluded that the quality and level of detail of the Plant A FPIE and internal flooding PRA, as well as FPRA, was sufficient to support the categorization of SSCs. The NRC staff's previous review of and finding on the technical acceptability of the Plant A FPIE and FPRA is for the same purpose as that used in the case study, and therefore, is valid for this review.

Plant A – Key Assumptions and Sources of Uncertainty

In its letter dated July 19, 2019, the licensee identified key assumptions and sources of uncertainty and provided corresponding dispositions for the Plant A SPRA used for the case study. The NRC staff reviewed the information provided by the licensee. In addition, the NRC staff reviewed the sensitivities reported for the SPRA in the Plant A response to the 10 CFR 50.54(f) letter and the corresponding NRC staff's response letter to the licensee for Plant A (Reference 33). The NRC staff's response letter for the Plant A SPRA submitted for the

10 CFR 50.54(f) letter states that complete elimination of FLEX credit from the SPRA resulted in an insignificant impact on the results. In its July 19, 2019, letter, the licensee dispositioned the core cooling success criteria following containment failure or venting based on an EPRI guidance document for development of SPRAs. The NRC staff has not reviewed or endorsed the EPRI guidance document for licensing activities. However, the NRC staff's review of the sensitivities reported for the SPRA in the Plant A response to the 10 CFR 50.54(f) letter determined that changing the core cooling success criteria following containment failure or venting has an insignificant impact on the SPRA results. Therefore, the NRC staff finds that the core cooling success criteria following containment failure or venting is adequately dispositioned for the Plant A case study.

The NRC staff's review of the sensitivities reported for the SPRA in the Plant A response to the 10 CFR 50.54(f) letter determined that a bounding sensitivity eliminating any credit for operator actions to reset relays is not expected to change the conclusions from the Plant A case study because the sensitivity will make relay chatter more important and relay chatter is identified as a categorization conclusion in EPRI Report 3002012988. The NRC staff further finds that the remaining assumptions and sources of uncertainty identified for the Plant A SPRA in the July 19, 2019, supplement are not key for the Plant A case study when evaluated with respect to the definitions of key assumption and sources of uncertainty in RG 1.200, Revision 2.

In the licensee's letter dated July 19, 2019, key assumptions and sources of uncertainty for Plant A FPIE and FPRA identified in the Plant A 10 CFR 50.69 LAR (Reference 34) were determined to be applicable to the Plant A case study supporting the alternative seismic approach. The NRC staff's previous review of the Plant A FPIE and FPRA for Plant A's 10 CFR 50.69 LAR concluded that the licensee searched for, identified, and resolved sources of uncertainty in its FPIE and FPRA consistent with the relevant guidance, except for the implementation items related to the Plant A FPIE and FPRA. The NRC staff's previous review of key assumptions and sources of uncertainty from the Plant A FPIE and FPRA on categorization of SSCs is for the same purpose as that used in the case study, and therefore, is valid for this review.

The NRC staff's evaluation for the Plant A 10 CFR 50.69 LAR resulted in four implementation items for the Plant A FPIE and internal flooding PRA. Similarly, seven implementation items were identified for the Plant A FPRA. The NRC staff reviewed the implementation items for the Plant A FPIE, internal flooding, and FPRA to determine whether the implementation items had the ability to impact the conclusions from the Plant A case study.

In the licensee's letter dated July 19, 2019, implementation items in the Plant A FPIE and FPRA were identified as additional key assumptions and sources of uncertainty relevant to the Plant A case study in EPRI Report 3002012988. The sensitivity study that removed credit for core melt arrest in-vessel at high reactor pressure vessel pressure did not result in any changes to the categorization conclusions from the case study. A separate sensitivity study for the emergency diesel generator (EDG) cooling fan success criteria revealed that the four EDG cooling fans would be categorized as HSS from the SPRA due to the correlated failure mode of those fans. In addition, a sensitivity study to address all four implementation items for the Plant A FPIE, which was performed by making changes to the Plant A FPIE and SPRA, also resulted in four EDG cooling fans categorized as HSS due to correlated failure mode. Results using the Plant A FPIE to model the EDG cooling fan success criteria showed that the fans would be HSS from FPIE. The licensee explained that the core damage DID and shutdown risk criteria in the categorization process in NEI 00-04 would also result in the four EDG fans being categorized as HSS. The licensee further explained that the sensitivity studies did not result in any new HSS

categorization from the SPRA, which was not captured by the FPIE, and therefore, the implementation items related to the FPRA, which only impact the results of the FPRA, would not change the categorization conclusions from the Plant A case study.

The NRC staff finds that the key assumptions and sources of uncertainty in the Plant A SPRA, FPIE, and FPRA resulted in additional HSS components due to correlated failure, which is a failure mode already included as a categorization conclusion in EPRI Report 3002012988. The NRC staff also finds that the implementation item for the Plant A internal flooding PRA does not impact the case study. Therefore, the NRC staff finds that key assumptions and sources of uncertainty in the Plant A SPRA, FPIE, and FPRA are dispositioned for the Plant A case study.

Plant A - Technical Acceptability Conclusion for PRAs Used in the Plant A Case Study

Based on its review, the NRC staff finds the Plant A SPRA, FPIE, and FPRA to be technically acceptable for use in the Plant A case study supporting the proposed alternative seismic approach because:

1. The SPRA was peer-reviewed following NRC-accepted guidance, and the results of the peer review were available to the NRC staff.

2. The consideration of the differences between the SRs in Part 5 of the 2013 ASME/ANS PRA Standard and in the 2009 ASME/ANS PRA Standard in the review of the technical acceptability of Plant A SPRA for the case study supporting the proposed alternative seismic approach does not impact the technical acceptability of the Plant A SPRA for the case study.

3. All finding level F&Os from the Plant A SPRA peer review were dispositioned based on the potential impact of the findings on the conclusions from the Plant A case study.

4. The NRC staff's evaluation for the Plant A 10 CFR 50.69 LAR found the FPIE and FPRA to be technically acceptable and that the conclusion is valid for this review because the FPIE and FPRA are being used for categorization related conclusions.

5. The impact of the implementation items from the Plant A 10 CFR 50.69 LAR either did not impact the categorization conclusions from the Plant A case study or are included in the proposed alternative seismic approach.

6. Key assumptions and sources of uncertainty in the Plant A SPRA, FPIE, and FPRA were identified, and those assumptions either did not impact the conclusions from the Plant A case study, or the impacts are captured in the proposed alternative seismic approach.

Plant C - Peer Review Process and Resolution of Peer Review Findings

Section 3.4.1 of EPRI Report 3002012988 provided information on the peer review of the SPRA, FPIE, and FPRA used for the Plant C case study. According to the information in Section 3.4.1 of EPRI Report 3002012988, the SPRA for Plant C used for the case study was peer-reviewed in November 2014 and has been revised to address the resulting F&Os. Additional information regarding the peer review of the Plant C SPRA was available to the NRC staff from its evaluation of the LAR to incorporate the SPRA into an approved 10 CFR 50.69 categorization process for Plant C (Reference 35).

The NRC staff evaluated the technical acceptability of the licensee's SPRA as part of its review of the Plant C LAR to incorporate the SPRA into an approved 10 CFR 50.69 categorization process. The review included an evaluation of the licensee's resolution of the finding level F&Os from the peer review. The NRC staff also reviewed the difference between the SRs in Part 5 of the 2013 ASME/ANS PRA Standard against those in the 2009 ASME/ANS PRA Standard, which is endorsed by RG 1.200, Revision 2. The review also included an evaluation of the impact of open internal events finding level F&Os on the SPRA. The NRC staff's evaluation on the Plant C LAR to incorporate the SPRA into an approved 10 CFR 50.69 categorization process concluded that the quality and level of detail of the Plant C SPRA was sufficient to support the categorization of SSCs. The Plant C SPRA was previously reviewed for the same purpose as that in the case study, and therefore, the NRC staff's conclusion regarding the technical acceptability of the Plant C SPRA for the LAR to incorporate the SPRA into an approved 10 CFR 50.69 categorization process is valid for this review. The NRC staff, in its previous review of the Plant C SPRA for use in the 10 CFR 50.69 categorization, also concluded that the use of Part 5 of the 2013 ASME/ANS PRA Standard (also known as Addendum B) is an acceptable alternative to the NRC-endorsed approach for the Plant C SPRA used to support categorization of SSCs under 10 CFR 50.69 (Reference 35).

Section 3.4.1 of EPRI Report 3002012988 stated that the FPIE used for the Plant C case study underwent a full scope peer review in 2009 and has been revised to resolve all F&Os received during the peer review. The Plant C FPRA used for the case study underwent a full scope peer review in 2012 and has been revised to resolve all F&Os received during the peer review. The NRC staff's evaluation on the Plant C LAR to adopt 10 CFR 50.69 concluded that the technical acceptability of the Plant C FPIE (including internal flooding) and FPRA was sufficient to support that application (Reference 36). The NRC staff's conclusion on Plant C FPIE and FPRA for adoption of 10 CFR 50.69 was based on the corresponding base PRAs. In addition, the NRC staff found Plant C FPIE to be acceptable in its review of the Plant C LAR to incorporate the Tornado Missile Risk Evaluator (TMRE) methodology in its licensing basis (Reference 37). The categorization of SSCs also uses the base PRAs. Therefore, the NRC staff's conclusion regarding the technical acceptability of the Plant C FPRA for the adoption of 10 CFR 50.69 application, as well as the Plant C FPIE for the adoption of 10 CFR 50.69 and TMRE applications, is valid for this review.

The NRC staff's evaluations of the Plant C FPIE for the TMRE application and the Plant C SPRA for the LAR to incorporate the SPRA into an approved 10 CFR 50.69 categorization process document different approaches for modeling the reactor coolant pump low leakage seals (also known as shutdown seals or SDS). It was unclear to the NRC staff whether the different modeling approaches impacted the conclusions from the Plant C case study. In the licensee's letter dated July 19, 2019, the results of a sensitivity performed using consistent modeling of SDS across all Plant C PRAs used in the case study were discussed as being consistent with those from the Plant C case study in EPRI Report 3002012988. The NRC staff finds that the differences in modeling SDS between the Plant C PRAs used for the case study do not impact the corresponding categorization conclusions.

Plant C - Key Assumptions and Sources of Uncertainty

In its letter dated July 19, 2019, the licensee stated that the key assumptions and sources of uncertainty for Plant C FPIE, FPRA, and SPRA identified in the Plant C 10 CFR 50.69 LARs (Reference 38 and Reference 39) were applicable to the Plant C case study supporting the proposed alternative seismic approach. The licensee stated that no additional key assumptions

and sources of uncertainty for the Plant C FPIE, FPRA, and SPRA relevant to the case study were identified.

The NRC staff's previous review of the results of the Plant C FPIE, FPRA, and SPRA for technical acceptability for Plant C 10 CFR 50.69 LARs concluded that the licensee searched for, identified, and resolved sources of uncertainty in its FPIE, FPRA, and SPRA consistent with the relevant guidance. The NRC staff concluded that additional sensitivities related to disposition of key assumptions and sources of uncertainty were not required for Plant C FPIE, FPRA, and SPRA. The NRC staff's previous review of and finding on the resolution of key assumptions and sources of uncertainty from Plant C FPIE, FPRA, and SPRA on categorization of SSCs is for the same purpose as in the case study in EPRI Report 3002012988, and therefore, is valid for this review.

Plant C - Technical Acceptability Conclusion for PRAs Used in the Plant C Case Study

Based on its review, the NRC staff finds the Plant C SPRA, FPIE, and FPRA to be technically acceptable for use in the Plant C case study supporting the proposed alternative seismic approach because:

1. The NRC staff's evaluation for the Plant C LAR to incorporate its SPRA in an approved 10 CFR 50.69 program found the SPRA to be technically acceptable and that the conclusion is valid for this review because the SPRA is being used for categorization related conclusions.

2. The NRC staff's evaluation for Plant C's application for adoption of 10 CFR 50.69 found the FPRA to be technically acceptable for that application and that the conclusion is valid for this review because the base FPRA is being used in the case study.

3. The NRC staff's evaluation for Plant C's applications for adoption of 10 CFR 50.69 and TMRE found the FPIE to be technically acceptable and that the conclusion is valid for this review because the base FPIE is being used in the case study.

4. It was demonstrated that the difference in modeling the low leakage reactor coolant pump seals in the different Plant C PRAs does not impact the categorization conclusions from the Plant C case study.

5. Key assumptions and sources of uncertainty in the Plant C SPRA, FPIE, and FPRA were identified, and those assumptions did not impact the conclusions from the Plant C case study.

Plant D - Peer Review Process and Resolution of Peer Review Findings

EPRI Report 3002012988 did not include any information on the peer review of the FPIE and SPRA used for the Plant D case study. Section 3.5.4 of EPRI Report 3002012988 stated that Plant D did not have an FPRA.

Details regarding the peer review of the SPRA for Plant D were available to the NRC staff from publicly available information submitted as part of the response for Plant D to the 10 CFR 50.54(f) request arising from NTTF recommendation 2.1 (References 28 and 29), which were incorporated by reference by the licensee in its letter dated May 10, 2019. Additional information about the Plant D SPRA was available to the NRC staff from publicly available

information submitted as part of the Plant D 10 CFR 50.69 LAR (Reference 30), which was also incorporated by reference by the licensee in the May 10, 2019, supplement.

Appendix A of the Plant D response to the 10 CFR 50.54(f) request contains detailed information on the peer review for the Plant D SPRA, the F&O closure review for the SPRA, and the open finding level F&Os, as well as their corresponding dispositions. The Plant D SPRA peer review was performed using the guidance in NEI 12-13 and the requirements in Part 5 of the 2013 ASME/ANS PRA Standard. RG 1.200, Revision 2, endorses the 2009 ASME/ANS PRA Standard and does not endorse the 2013 ASME/ANS PRA Standard. As part of the review of Plant D response to the 10 CFR 50.54(f), the NRC staff found that the F&O closure process was appropriately implemented for the Plant D SPRA and that the open SPRA finding level F&O, as well as the FPIE finding level F&Os did not impact the SPRA. The NRC staff has previously reviewed the differences between the SRs in Part 5 of the 2013 ASME/ANS PRA Standard against those in Part 5 in the 2009 ASME/ANS PRA Standard and found the use of Part 5 of 2013 ASME/ANS PRA Standard (also known as Addendum B) to be an acceptable alternative to the NRC-endorsed approach for the Plant C SPRA used to support categorization of SSCs under 10 CFR 50.69.

In addition, the NRC staff also reviewed SRs in Part 5 of the 2013 ASME/ANS PRA Standard that can have plant-specific differences using the information available for the Plant D SPRA. Therefore, the NRC staff included consideration of the differences between the SRs in Part 5 of the 2013 ASME/ANS PRA Standard and in the 2009 ASME/ANS PRA Standard in the review of the technical acceptability of Plant D SPRA for the case study supporting the proposed alternative seismic approach. The purpose of determining the technical acceptability of Plant D SPRA for this application is to determine the appropriateness of the conclusions from the Plant D case study and not for a Plant D-specific licensing action. Based on the information available to the NRC staff, as well as its review, the NRC staff finds that use of Part 5 of the 2013 ASME/ANS PRA Standard does not impact the technical acceptability of the Plant D SPRA for use in the case study.

Information about the peer review of the Plant D FPIE was available to the NRC staff from publicly available information submitted as part of the Plant D 10 CFR 50.69 application. Section 3.3 of the Plant D 10 CFR 50.69 application states that the Plant D FPIE (with internal flooding) was subjected to a self-assessment and a full-scope peer review in November 2009 following the guidance in RG 1.200, Revision 2. Further, a finding closure was also conducted in July 2017 for the FPIE, as described in EPRI Report 3002012988. Attachment 3 of the Plant D 10 CFR 50.69 application provides a discussion of the open finding level F&Os for the FPIE and their corresponding dispositions for the 10 CFR 50.69 application. In its letter dated July 19, 2019, the licensee discussed the impact of the open SPRA and FPIE finding level F&Os on the conclusions from the Plant D case study. The licensee stated that there were seven open finding level F&Os for the FPIE and one open finding level F&O for the SPRA for Plant D. The licensee further explained that four of the eight open finding level F&Os dealt with uncertainty analysis or were documentation issues and did not impact the case study. As noted previously, the staff reviewed the open finding level F&O for the Plant D SPRA and determined that it did not impact that SPRA. In its letter dated July 19, 2019, the licensee provided additional details to justify the disposition of one open finding level F&O and discussed the results of sensitivity studies performed for three open finding level F&Os, which did not change the conclusions from the Plant D case study.

The NRC staff reviewed the open finding level F&Os for the Plant D FPIE and their corresponding dispositions in the Plant D 10 CFR 50.69 application, as well as the information

provided in the July 19, 2019 letter. The NRC staff finds that that the open FPIE finding level F&Os do not impact the conclusions from the Plant D case study.

Only permanently installed FLEX components were credited in the Plant D FPIE and SPRA. The failure probabilities used for the FLEX equipment were taken from similar type components (e.g., the FLEX diesel generators used the EDG data). In its letter dated July 19, 2019, the licensee discussed the results of a sensitivity by increasing the failure probabilities of FLEX components to 3 times the original value which did not produce any new categorization conclusions compared to the case study.

The NRC staff reviewed the information related to the modeling of FLEX equipment. The NRC staff's review also included additional details provided in the Plant D response to the 10 CFR 50.54(f) letter on the FLEX equipment and operator actions that were credited in the SPRA, such as the inclusion of random and seismic-induced failures of such equipment. Further, the supplement to the Plant D response to the 10 CFR 50.54(f) letter discussed the determination of the FLEX component fragilities and the minor impact of variation in those fragilities on the SPRA results. Based on its review, the NRC staff finds that the modeling of FLEX equipment in the Plant D FPIE and SPRA is acceptable for the Plant D case study because changes to the modeling do not impact the categorization conclusions from the case study. The NRC staff notes that EPRI Report 3002012988 includes identification of FLEX equipment as HSS from SPRAs as a categorization conclusion from the case studies.

Plant D - Key Assumptions and Sources of Uncertainty

In its letter dated July 19, 2019, the licensee identified key assumptions and sources of uncertainty for Plant D FPIE and SPRA identified in the Plant D 10 CFR 50.69 LAR that were determined to be applicable to the Plant D case study. In addition, the licensee stated that the Plant D case study conclusions were not impacted by the key sources of uncertainty identified in the Plant D 10 CFR 50.69 LAR. The licensee also stated that no additional key assumptions and sources of uncertainty for Plant D FPIE and SPRA relevant to the case study were identified.

The NRC staff reviewed the information provided by the licensee, the sensitivities reported for the SPRA in the Plant D response to the 10 CFR 50.54(f) letter, and the corresponding NRC staff's response letter (Reference 40). The NRC staff's review of the sensitivities reported for the SPRA in the Plant D response to the 10 CFR 50.54(f) letter finds that all sensitivities except one have an insignificant impact on the SPRA results. The NRC staff's review of the sensitivity that eliminates correlated seismically-induced failures of SSCs in the SPRA, along with the revised Table 3-11 of EPRI Report 3002012988 submitted in the letter dated July 19, 2019, finds that the assumption of correlated failures is not expected to result in new conclusions from the Plant D case study. This staff finding is because the approach used for correlated failures is based on state of practice, and therefore, is not a key assumption for the Plant D case study based on the definitions of key assumption and sources of uncertainty in RG 1.200, Revision 2. Further, the sensitivity would result in individual relay-chatter failures becoming more important, and relay chatter is already identified as a categorization conclusion from SPRAs in EPRI Report 3002012988.

Plant D - Technical Acceptability Conclusion for PRAs Used in the Plant D Case Study

Based on its review, the NRC staff finds the Plant D SPRA and FPIE to be technically acceptable for use in the Plant D case study supporting the proposed alternative seismic approach because:

1. The NRC staff's evaluation of portions of the Plant D 10 CFR 50.69 LAR relevant to the proposed alternative seismic approach found the SPRA to be peer-reviewed following NRC-accepted guidance, and the results of the peer review were available to the NRC staff.

2. The NRC staff's evaluation of portions of the Plant D 10 CFR 50.69 LAR relevant to the proposed alternative seismic approach found the peer review of the FPIE to be adequate to support the case study, and the results of the peer review were available to the NRC staff.

3. The consideration of the differences between the SRs in Part 5 of the 2013 ASME/ANS PRA Standard and in the 2009 ASME/ANS PRA Standard in the review of the technical acceptability of Plant D SPRA for the case study supporting the proposed alternative seismic approach does not impact the technical acceptability of the Plant D SPRA for use in the case study supporting this application.

4. All finding level F&Os from the Plant D SPRA peer review were dispositioned with respect to the categorization conclusions from the case study based on the potential impact of the findings on the conclusions.

5. Key assumptions and sources of uncertainty in the Plant D SPRA and FPIE were identified, and those assumptions did not impact the conclusions from the Plant D case study.

Evaluation of Acceptability of Changes to the PRAs Used for Case Studies Supporting the Proposed Alternative Seismic Approach

Mapping of HSS SSCs between SPRA and FPIE, as well as FPRA, was an important aspect of the categorization conclusions from the Plant A, C, and D case studies. The mapping performed for Plant A, C, and D is discussed in Sections 3.2, 3.4, and 3.5 of EPRI Report 3002012988. The mapping was performed to, if possible, assign SSCs and SSC failure modes from the SPRA to representative SSCs and SSC failure modes in the FPIE or FPRA. The purpose of the mapping was to identify SSCs and SSC failure modes from the SPRA that were not captured in the FPIE or, as applicable, FPRA.

The NRC staff reviewed the approach, implementation, and results of the mapping performed for the Plant A, C, and D case studies to ensure that the mapping was technically justified, and therefore, could be used to develop the categorization conclusions from the case studies. The NRC staff's review included the details in Section 3.2.5 and the results in Table 3-5 of EPRI Report 3002012988 for Plant A, the details in Section 3.4.51 and the results in Table 3-9 of EPRI Report 3002012988 for Plant C, and the details in Section 3.5.5 and the results in Table 3-11 of EPRI Report 3002012988 for Plant D.

As part of the mapping, each SSC that belonged to a seismic failure event (fragility group) in the SPRA that exceeded the HSS thresholds in NEI 00-04 was evaluated, as described in EPRI Report 3002012988, to determine if the same HSS determination would be made from either the FPIE PRA or FPRA. Some of the seismic failure events and the corresponding SSCs identified as being HSS from the SPRA were explicitly modeled in either the FPIE or FPRA and

could be directly mapped. However, for cases where such explicit modeling was not present in either the FPIE or FPRA, mapping to a representative basic event in the FPIE or FPRA was performed, as described in EPRI Report 3002012988. The representative basic events included so-called 'super components' (i.e., components which are either impacted by or include the seismic failure event of interest) or operator actions. The mapping of the implicitly modeled components for each of the seismic failure events of interest was discussed in Sections 3.2.5.2, 3.4.5.2, and 3.5.5.2 of EPRI Report 3002012988 for Plants A, C, and D, respectively.

EPRI Report 3002012988 did not include a discussion of the approach used for combining importance measures for seismically-induced and random failures to generate the final importance measure for use in developing the categorization conclusions.

In its letter dated July 19, 2019, the licensee provided information of the development of the importance measures from the SPRA for use in the case studies for Plants A, C, and D. Plants A, C, and D did not combine the importance measures for seismically-induced and random failures as part of the respective case studies. The licensee stated that a more detailed review performed for Plant A by combining the importance measures for seismically-induced and random failures did not result in any additional SSCs becoming HSS from the SPRA compared to the case study. The evaluation of the impact of combining the importance for seismically-induced and random failures for Plant C did not show additional SSCs being identified as HSS compared to the Plant C case study. In its letter dated July 19, 2019, the licensee provided a revision to Table 3-9 of EPRI Report 3002012988 to clarify certain entries that were HSS only due to random failure modes for the Plant C case study. A sensitivity study to determine the impact of combining the importance for seismically-induced and random failures for Plant D resulted in some additional components being categorized as HSS from the SPRA due to the contribution of the random failures. In addition, the revision to Table 3-11 of EPRI Report 3002012988, provided in letter dated July 19, 2019, included the additional components. The licensee explained that the additional components that were HSS uniquely from the SPRA would be identified as HSS by other parts of the categorization process discussed in NEI 00-04, such as DID. In its letter dated July 19, 2019, the licensee provided an addition to EPRI Report 3002012988, as Section 3.6.5 to that report, to discuss the ability of DID considerations in NEI 00-04 to assure that DID is preserved when categorizing an SSC as LSS. The NRC staff's review of the impact of combining seismically-induced and random failures finds that such combinations either did not impact the case studies or are included in the conclusions in EPRI Report 3002012988, including the addition, and therefore, in the proposed alternative seismic approach.

In several cases, passive components, such as tanks, were mapped to operator actions, such as those involving manipulation of valves to align the valves to the tank. An example of such mapping included the condensate storage tank (CST) for Plant A. Categorization following the guidance in NEI 00-04 is performed on a component basis, and therefore, it was unclear to the NRC staff whether such mapping was justified. In its letter dated July 19, 2019, the licensee explained that the mapping of the CST was appropriate because failure of the valve or the CST itself would fail the action of refilling the CST inventory. As a result, even if the mapping were not performed, the CST would still be categorized as HSS by the function associated to the valve, as well as other aspects of categorization such as DID. An explanation of the mapping from the SPRA to the FPIE or FPRA for the Plant C case study was included in the letter dated July 19, 2019. Plant D did not map any passive components to operator actions. The NRC staff's review of the information presented in the licensee's letter dated July 19, 2019, and EPRI Report 3002012988 finds that mapping between distinct components in the case studies for Plants A, C, and D is either not performed or is justified.

The discussion in Tables 3-8 and 3-10 of EPRI Report 3002012988 for the mapping of passive or implicitly modeled SSCs for Plants C and D indicated that building failures were mapped to basic events in the FPIE that represented failure of the SSCs within the building, typically the common cause failure of the SSCs. However, the NRC staff's review of Tables 3-9 and 3-11 of EPRI Report 3002012988 indicated that building failures were not HSS, and therefore, did not need to be mapped to any SSCs in the FPIE. In its letter dated July 19, 2019, the licensee clarified that seismically-induced building failure events in the SPRA did not meet HSS criteria established in NEI 00-04 for Plant A and D case studies. In its letter dated August 5, 2019, the licensee clarified its response on seismically-induced building failures for the Plant A case study to reiterate that mapping of building failures was not performed for the Plant A case study because none of the building failures met the criteria for HSS. In its letter dated July 19, 2019, the licensee provided a revision to Table 3-10 of EPRI Report 3002012988 that removed the discussion of building failures from that table since the discussion was not applicable to Plant D. Plant C mapped building failures that were HSS to the most obvious component(s) within the building and provided relevant examples of such mapping. In its letter dated July 19, 2019, the licensee clarified that the mapping approach for building failures used by Plant C did not change the importance measures of the mapped and unmapped components. Exelon provided a revision to Table 3-8 of EPRI Report 3002012988 to identify such building failures that were HSS from the SPRA.

Exelon also provided Section 3.6.6 as an addition to EPRI Report 3002012988 to discuss categorization of civil structures. Section 3.6.6 to EPRI Report 3002012988 states that civil structures containing PRA credited equipment (e.g., reactor building) are likely important to safety because their failure can fail the credited equipment functions and recommends considering civil structures housing HSS SSCs to be HSS themselves, unless otherwise justified, in the event of a licensee choosing to categorize structures under the 10 CFR 50.69 program. The NRC staff's review of the mapping of building failures in the case studies finds that the mapping identifies the seismically-induced structural failure of buildings and the impact of such failures on SSCs within the structure. The NRC staff also finds that the proposed alternative seismic approach includes appropriate consideration of building failures by recommending the structures housing HSS SSCs to be HSS themselves. The NRC staff notes that it did not review and is not providing any conclusions on the discussion related to the appropriateness of the use of the risk achievement worth importance measure for SPRAs in Section 3.6.6, which was provided as an addition to EPRI Report 3002012988 by Exelon for the proposed alternative seismic approach.

Based on the discussion in Table 3-8 of EPRI Report 3002012988 for the Plant C case study, it was unclear how such mapping could capture the safety significance of the impacted SSCs. In its letter dated July 19, 2019, the licensee explained that failure of containment penetrations, such as electrical and mechanical penetrations, was modeled to represent failure of containment, and that seismically-induced failure of containment isolation valves was mapped to basic events in the containment isolation fault tree in the FPIE and not the end state. Seismically-induced containment penetration failure events in the SPRA did not meet HSS criteria established in NEI 00-04 for Plant A, and therefore, mapping of such failure events was unnecessary for Plant A. Failure of the containment penetration fragility group in the Plant D SPRA was modeled to lead directly to a large containment isolation failure, and the containment penetration group was found to be HSS from the SPRA, which would result in all penetrations being categorized as HSS. Exelon provided a revision to Table 3-11 of EPRI Report 3002012988 to include an entry for containment penetrations. In addition, the revision to Table 3-11 of EPRI Report 3002012988 states that DID considerations in the NEI 00-04 categorization process would also result in the categorization of containment penetrations as

HSS. As noted previously, the licensee provided Section 3.6.5 as an addition to EPRI Report 3002012988 to discuss the ability of DID considerations in NEI 00-04 to assure that DID is preserved when categorizing an SSC as low safety-significant. The NRC staff's review of the impact of mapping containment penetration failures finds that such failures are included in the categorization conclusions in EPRI Report 3002012988, including the addition of Section 3.6.5, and therefore, in the proposed alternative seismic approach.

Based on its review of the information in EPRI Report 3002012988, specifically Sections 3.2.5.2, 3.4.5.2, and 3.5.5.2; Tables 3-5, 3-9, and 3-11; and the information in the licensee's letter dated July 19, 2019, the NRC staff finds that the mapping of SSCs between the SPRA, FPIE and, as applicable, FPRA for Plant A, C, and D case studies, was performed in a technically justifiable manner because (1) the mapping of explicitly modeled components was performed by identifying representative or logically related SSCs, including so-called 'super components' among the PRAs, (2) the mapping of passive and implicitly modeled components was performed by identifying appropriate functions for such components, and (3) the building and containment penetration failures were included in the categorization conclusions derived from the case studies in EPRI Report 3002012988, and therefore, in the proposed alternative seismic approach.

Evaluation of the Conclusions from the Case Studies Supporting the Proposed Alternative Seismic Approach

Section 3.6 of EPRI Report 3002012988 discusses the conclusions on the determination of unique HSS SSCs from SPRAs to support the proposed alternative seismic approach.

The key categorization conclusion from the Plant A, C, and D case studies is that the only SSCs identified as HSS in the SPRA, that were not also HSS from other assessments that are part of 10 CFR 50.69 categorization, including FPIE and/or FPRA, were from unique seismically-induced failure modes. The remainder of HSS SSCs from SPRA are captured by the corresponding FPIE, FPRA, or other aspects of the NEI 00-04 categorization process. Additional details about the categorization conclusions is provided in Sections 3.6.2 through 3.6.4 of EPRI Report 3002012988.

Section 3.6.2 of EPRI Report 3002012988 states that the case studies identified HSS SSCs unique to SPRAs due to the correlated failure mode of those SSCs during a seismic event (e.g., if two pumps performing the same function are located side by side in a plant, they are both assumed to fail with the same conditional probability of failure for a given seismic acceleration). Section 3.6.3 of EPRI Report 3002012988 discusses the conclusions from the case studies related to relays and states that relays are explicitly modeled in SPRAs but are usually not included in FPIE. EPRI Report 3002012988 states that relays would be implicitly modeled in the FPIE PRA and their function within the system would need to be evaluated to perform 10 CFR 50.69 categorization down to the component level based on the guidance in Section 5 of NEI 00-04 for implicitly modeled components.

The NRC staff reviewed the results of the case studies presented in Sections 3.2.5, 3.4.5, and 3.5.5 of EPRI Report 3002012988 for Plants A, C, and D, respectively. The NRC staff evaluated unique aspects of SPRAs as part of its review of the conclusions on the determination of unique HSS SSCs from SPRAs to support the proposed alternative seismic approach. Appendix 5-A of the 2009 ASME/ANS PRA Standard discusses the important differences between SPRA and FPIE, including:

- Initiating events caused by earthquakes that can be different from those in FPIE;
- Consideration of the entire seismic hazard curve (i.e., all possible levels of earthquakes along with their frequencies of occurrence and consequences); and
- Simultaneous damage to multiple redundant components (also known as correlated failures), which represents a major common-cause effect.

Additional discussion in the 2009 ASME/ANS PRA Standard notes that earthquakes can cause failures that are not explicitly represented in the FPIE such as damage to structures and other passive items such as large tanks and anchorage. Other categories of seismically-induced failures that are typically not modeled in the FPIE are seismically-induced relay chatter. In addition, the 2009 ASME/ANS PRA Standard also notes that SPRAs need to consider the physical locations and proximity of SSCs because of secondary failures such as spatial interactions. These differences and special considerations for SPRA, as compared to FPIE, are consistent with the discussion in NUREG/CR-2300, Volume 2, "PRA Procedures Guide: A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants: Chapters 9-13 and Appendices A–G" (Reference 41).

The NRC staff's review of the results of the case studies for Plants A, C, and D, notes that the SSCs that were identified as HSS uniquely from the SPRAs are primarily due to the correlated failure mode of SSCs, failure of passive structures, and the relay-chatter failure mode. The NRC staff finds that SSCs experiencing such failure modes are expected to be identified as HSS from SPRAs because these failure modes are unique to, and dominant in, SPRAs. The NRC staff further finds that the identification of SSCs as HSS due to such failure modes uniquely from SPRAs is consistent with the understanding of the impact of seismic events on nuclear power plants, as well as the development of SPRAs.

The NRC staff's evaluation of the categorization conclusion related to relays due to relay-chatter failure mode determined that subcomponents such as relays that are not directly modeled in other PRAs could be treated as another failure mode for the SSCs to which they are associated. Therefore, the importance of relays would be accounted for in the importance calculation for the corresponding SSCs using the NEI 00-04 formulae for the integral assessment. The NRC staff's review notes that although EPRI Report 3002012988 appears to focus on emergency power system relays, the categorization conclusion for relays is applicable to any relay modeled in the SPRA because the relays being modeled in the SPRA implies that those relays are important for the function of an SSC.

Section 3.6.4 of EPRI Report 3002012988 discusses the categorization conclusion related to identification of FLEX equipment from the case studies. Plant A and D modeled FLEX equipment in their SPRA or FPIE, and such equipment is identified as HSS either uniquely from the SPRA or from the SPRA and FPIE PRA both. The discussion in Section 3.6.4 of EPRI Report 3002012988 appears to indicate that meeting PRA technical acceptability through the guidance in RG 1.200, Revision 2, as well as the relevant ASME/ANS PRA Standard ensures that the performance assumed in the PRA for FLEX equipment is consistent with plant practices. The NRC staff disagrees with that interpretation of the guidance in RG 1.200, Revision 2, because the guidance in RG 1.200, Revision 2, ensures that SSC reliability data used in PRAs reflect the as-built, as-operated plants, and the guidance does not support maintaining the assumed performance of such SSCs, which is required by 10 CFR 50.69(e). Nonetheless, the interpretation in EPRI Report 3002012988 on the use of PRA technical acceptability guidance in RG 1.200, Revision 2, for performance monitoring for FLEX equipment

does not change the categorization conclusions about unique HSS SSCs from SPRAs, or the licensee's implementation in the proposed alternative seismic approach for Calvert Cliffs, and therefore, does not impact the NRC staff's decision.

Based on its review, the NRC staff finds that the conclusions on the determination of unique HSS SSCs from SPRAs in EPRI Report 3002012988 from the Plant A, C, and D case studies and Exelon's application are valid because (1) they were developed following a systematic process that used technically acceptable SPRAs; (2) the HSS SSCs from the SPRA, FPIE, and as applicable, FPRA were identified consistent with the guidance in NEI 00-04; (3) mapping of SSCs between the PRAs used in each case study was performed in a technically justifiable manner to identify conclusions; (4) the categorization conclusions were consistent with the SSCs identified as uniquely HSS from SPRAs in the Plant A, C, and D case studies; and (5) the categorization conclusions are consistent with the unique aspects and key differences of SPRAs.

Evaluation of the Criteria for the Proposed Alternative Seismic Approach

Exelon proposed the following criteria for the applicability and use of the proposed alternative seismic approach:

> GMRS peak acceleration is at or below approximately 0.2g or where the GMRS is below or approximately equal to the SSE between 1.0 Hz and 10 Hz.

The NRC staff reviewed the licensee's proposed criteria for its acceptability to determine the applicability and use of the proposed alternative seismic approach by the licensee.

For the currently operating plants, the SSE was developed to envelope the deterministic hazard at the site. Therefore, from a seismic hazard perspective, the site-specific SSE derived using a deterministic approach can be compared to the corresponding GMRS, which is derived using a probabilistic approach. The comparison of the site-specific GMRS and SSE provides information about any seismic risk that would be unaccounted for in the current plant licensing basis.

According to NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities" (Reference 42)., frequencies under 10 Hz are more closely related to the types of motion that could cause damage at nuclear power plants. EPRI Report 1025287 (Reference 43), which was endorsed by the NRC in letter dated November 15, 2013 (Reference 44), also focuses on the 1 to 10 Hz frequency range and states that empirical evidence, including earthquake data at nuclear power plants, demonstrates that high frequency vibratory motion during earthquakes is not damaging to nuclear power plant SSCs. The criteria for determination of exceedance of the operating basis earthquake at nuclear power plants in RG 1.166, "Pre-Earthquake Planning and Immediate Nuclear Power Plant Operator Post-Earthquake Actions" (Reference 45), focuses on the less than 10 Hz range. Exceptions to the susceptibility of SSCs in nuclear power plants in the 1 to 10 Hz range are the functional performance of vibration sensitive components, such as relays and other electrical and instrumentation devices. The case studies supporting this application are based on plants where the GMRS exceeds the SSE by a significant amount. The case studies also used SPRAs where the full range of the seismic hazard, including the high frequency portion, is included. The categorization conclusions from the case studies evaluated components susceptible of high frequency excitation, such as relays. The proposed alternative seismic approach includes explicit consideration of such components.

The seismic fragility of an SSC is based on the site-specific load demand on the SSC during a seismic event and the SSC's capacity to accommodate that load. SSCs have inherent capacities to accommodate seismic loading based on the non-seismic design loads, such as those from pressure, temperature, and dead weight, as well as the required functions for the SSC, irrespective of the site-specific seismic load. In addition, conservatisms in the design process result in margins above the design basis for such SSCs. Certain features such as equipment anchorage are designed against the site-specific seismic demand, and therefore, are more closely associated with the site-specific seismic loading. However, the design-basis criteria for such features include conservatisms that introduce margins to failure of such features, and consequently, the corresponding SSCs to perform their function. As stated in EPRI Report 1025287, experience has shown a low likelihood of a seismically-designed SSC being damaged by ground motions with a GMRS peak below 0.4g in the 1 to 10 Hz range. In addition, the IPEEE high confidence of low probability of failure (HCLPF) for Calvert Cliffs is approximately 0.3g (anchored to the peak ground acceleration). The HCLPF represents the estimate for which there is 95 percent confidence that the conditional failure probability is 5 percent or less. Therefore, there is a low likelihood of SSCs being damaged by accelerations at or below 0.3g at Calvert Cliffs.

Based on its evaluation, the NRC staff finds that the licensee's proposed criteria of GMRS peak acceleration is at or below approximately 0.2g or where the GMRS is below or approximately equal to the SSE between 1.0 and 10 Hz, to determine the applicability and use of the proposed alternative seismic approach, is acceptable because the criteria (1) provide information about any seismic risk that would be unaccounted for in the current plant licensing basis, (2) include the seismic acceleration in the frequency range of 1 to 10 Hz where a wide range of nuclear power plant SSCs are susceptible to seismically-induced damage, and (3) provide sufficient margin to accelerations below which the likelihood of seismically-designed SSCs being damaged by ground motions is low. In addition, the proposed alternative seismic approach includes explicit consideration of components susceptible of high frequency excitation such as relays.

Evaluation of Applicability of Criteria for the Proposed Alternative Seismic Approach to Calvert Cliffs

Exelon compared the GMRS from the reevaluated seismic hazard for Calvert Cliffs developed in response to NTTF Recommendation 2.1 against the site's design-basis SSE to demonstrate that Calvert Cliffs meets the criteria for application of the proposed alternative seismic approach. The NRC staff previously evaluated the licensee's response to the 10 CFR 50.54(f) letter associated with NTTF Recommendation 2.1 in which the licensee submitted its reevaluated seismic hazard (Reference 46). The NRC staff's previous assessment of the licensee's reevaluated seismic hazard states (Reference 47) that the licensee's methodology was acceptable and that the GMRS determined using the reevaluated hazard adequately characterized the site. Since the same reevaluated hazard is used for comparison against the criteria for use of the proposed alternative seismic approach, the NRC staff's previous assessment on the reevaluated hazard is applicable to this review. The NRC staff's evaluation of the licensee's reevaluated seismic hazard determined that the GMRS peak acceleration is below approximately 0.2g. Based on its review, the NRC staff finds that the licensee's reevaluated hazard for Calvert Cliffs meets the criteria for the proposed alternative seismic approach. Further, the NRC staff's review revealed that the Calvert Cliffs reevaluated hazard exceeds its SSE by approximately 18 percent in the 6 – 10 Hz frequency range. Therefore, the reevaluated GMRS exceeds the SSE in the 1 to 10 Hz by a substantially lower amount than the

corresponding exceedance for the case study plants shown in Figures 3-1, 3-3, and 3-4 of EPRI Report 3002012988.

Attachment 1 of the supplement dated May 10, 2019, included details about the risk contribution of a seismic event at Calvert Cliffs using a 'plant' level HCLPF estimate and the reevaluated seismic hazard. The licensee stated that the seismic risk estimate used in its approved RICT program (Reference 48) was based on a conservative HCLPF estimate used for the Calvert Cliffs expedited seismic evaluation process (ESEP) in response to the 10 CFR 50.54(f) letter associated with NTTF Recommendation 2.1. The licensee identified several sources of conservatism in the HCLPF estimate used for the Calvert Cliffs ESEP. The licensee stated that it reevaluated the HCLPF for the components in the Calvert Cliffs ESEP study that had the potential to challenge the limiting HCLPF identified in the Calvert Cliffs IPEEE and subsequent evaluations. The licensee explained that the results of these new evaluations demonstrated that the contribution of seismic risk at Calvert Cliffs could be estimated based on a 'plant' HCLPF estimate of 0.27g. Using that estimate, the licensee reported that the seismic CDF was approximately 2 percent of the total plant CDF (i.e., 2 percent of the sum of the internal events and internal fire CDF values reported in Attachment 2 of the November 28, 2018, application, plus the seismic CDF estimate) applicable to either unit. Using the seismic CDF and the 0.1 seismic conditional large early release probability estimate used for the Calvert Cliffs RICT program, the licensee reported that the seismic LERF was approximately 3 percent of total plant LERF reported in the LAR, applicable to either unit.

The NRC staff reviewed the information presented by the licensee in the supplement related to the estimated seismic risk at Calvert Cliffs. The NRC staff determined that conservatisms exist in the HCLPF estimates for the Calvert Cliffs ESEP. These conservatisms include uniformly amplifying the in-structure response spectrum over the entire frequency range, not considering building dynamic properties and the spectral shape of the GMRS, and using the conservatively biased prescriptive inputs from EPRI NP-6041-SL, "A Methodology for Assessment of Nuclear Plant Seismic Margin, Revision 1," for the conservative deterministic failure margin method that was used to calculate the HCLPF. Therefore, the NRC staff finds that the use of the 'plant' HCLPF estimate provided in Attachment 1 to the licensee's letter dated May 10, 2019, and the resulting seismic CDF and LERF estimates for Calvert Cliffs used to support the use of the proposed alternative seismic approach are well supported. Based on its review, the NRC staff concludes that the seismic risk contribution for Calvert Cliffs would not solely result in an SSC being categorized as HSS.

In summary, the NRC staff finds that the Calvert Cliffs basis for applying the proposed alternative seismic approach to its site is acceptable because: (1) the reevaluated hazard for Calvert Cliffs meets the criteria for use of the proposed alternative seismic approach, and (2) the seismic risk contribution for Calvert Cliffs would not solely result in an SSC being categorized as HSS.

Evaluation of the Implementation of Conclusions from the Case Studies Supporting the Proposed Alternative Seismic Approach

The categorization conclusions from the case studies indicated that seismic-specific failure modes resulted in HSS categorization uniquely from SPRAs. Therefore, such seismic-specific failure modes, such as correlated failures, relay chatter, and passive component structural failure mode, can influence the categorization process. The NRC staff reviewed the proposed alternative seismic approach to evaluate whether the categorization-related conclusions from EPRI Report 3002012988 were appropriately included and implemented.

In Attachment 1 to the licensee's letter dated May 10, 2019, Exelon discussed enhancements to its proposed alternative seismic approach. The licensee stated that the proposed alternative seismic approach will include qualitative consideration of the mitigation capabilities of SSCs during seismically-induced events and seismic failure modes based on insights obtained from prior seismic evaluations performed for Calvert Cliffs. As an example of such considerations, the licensee stated that as part of the categorization team's preparation of the system categorization document (SCD) that is presented to the IDP, a section would be included that would summarize the identified plant seismic insights pertinent to the SSC being categorized. The licensee further explained that at several steps of the categorization process, the categorization team will consider the available seismic insights relative to the system being categorized and document its conclusions in the SCD. In addition, the IDP would be provided with the basis for the proposed alternative seismic approach, including the low seismic hazard for the plant and the criteria for use of the proposed alternative seismic approach.

Attachment 1 to the licensee's letter dated May 10, 2019, included Figure 3-1, which showed enhancements to the major steps of the categorization process through the addition of two steps (represented by four blocks in Figure 3.1) that highlighted the review and consideration of seismic insights in the categorization of SSCs. In addition, Table 3-1 in Attachment 1 included an explicit mention of the categorization evaluation for seismic hazard that would be performed at either the functional or component level.

The licensee explained that the categorization team would review available Calvert Cliffs plant-specific seismic information and other resources to identify plant-specific seismic insights relevant to the SSCs being categorized, such as:

- Impact of relay chatter
- Implications related to potential seismic interactions such as with block walls
- Seismic failures of passive SSCs such as tanks and heat exchangers
- Any known structural or anchorage issues with a particular SSC
- Components that are implicitly part of PRA-modeled functions (including relays)
- Components that may be subject to correlated failures

The licensee further explained that these insights would provide the IDP a means to consider potential impacts of seismic events in the categorization process. The licensee stated that the IDP could challenge, from a seismic perspective, any candidate LSS recommendation for any SSC if it believed there was basis for doing so, and that any decision by the IDP to downgrade preliminary HSS components to LSS would also consider the applicable seismic insights.

The licensee explained that sources of the insights related to seismic events would be prior plant-specific seismic evaluations such as the seismic hazard screening; spent fuel pool assessment; expedited seismic evaluation process, as well as the seismic high frequency evaluation performed for NTTF Recommendation 2.1; seismic walkdowns performed for NTTF Recommendation 2.3; and seismic mitigation strategy assessment performed for NTTF Recommendation 4.2.

The licensee also provided Section 3.6.6 as an addition to EPRI Report 3002012988 to discuss categorization of civil structures. Section 3.6.6 states that civil structures containing PRA-credited equipment (e.g., reactor building) are likely important to safety because their failure can fail the credited equipment functions and recommends considering civil structures

housing HSS SSCs to be HSS themselves, unless otherwise justified in the event of a licensee choosing to categorize structures under the 10 CFR 50.69 program.

All case studies supporting this application included a corresponding FPIE, but only two of those case studies used information from an FPRA. Based on the information in EPRI Report 3002012988, the NRC staff determined that SSCs identified as HSS from SPRAs that overlapped with SSCs identified as HSS from corresponding FPRAs represented about 21 percent of the HSS fragility groups for Plant C. In addition, one fragility group was identified as HSS only from FPRA for Plant A. The NRC staff notes that a fragility group includes multiple SSCs, resulting in a higher contribution of HSS categorization from FPRAs to the categorization conclusions from the case studies. The guidance in NEI 00-04, as endorsed in RG 1.201, Revision 1, maintains the HSS categorization for SSCs identified as such from the FPIE and allows the use of the integrated importance measure determination, as well as the IDP to change the categorization for SSCs categorized as "candidate" HSS from the FPRA. In Attachment 1 of the supplement dated July 1, 2019, the licensee clarified that for SSCs that were uniquely HSS from the FPRA but not HSS from FPIE, the categorization team would review design-basis functions of the SSC(s) during seismic events or functions credited for mitigation and prevention of severe accidents caused by seismic events. The results of the review would be presented to the IDP as additional qualitative inputs and described in the SCD. The licensee further clarified that the discussion with the IDP will focus on SSCs that are uniquely HSS from FPRA because such SSCs may not be categorized as HSS following the integrated importance measure determination. Therefore, the proposed alternative seismic approach includes the categorization conclusions from the FPRAs in the Plant A, C, and D case studies by presenting and discussing insights related to the impact of seismic events to the IDP when a change in categorization of an SSC identified as HSS from the FPRA but not from the FPIE is possible.

According to the NEI 00-04 categorization process, once a component is categorized as HSS for a function, all other components within a system supporting that function are also initially assigned as HSS. The guidance in Section 5 of NEI 00-04 is for implicitly modeled components where such components are addressed by focusing on the function of the SSC being categorized under 10 CFR 50.69. The proposed alternative seismic approach will result in consideration of relays as implicitly modeled components and of insights related to the impact of seismically-induced relay chatter for the function achieved by the SSC during the categorization. Attachment 1 to the licensee's letter dated May 10, 2019, explicitly includes such insights in the information provided to the IDP as part of the categorization process. Therefore, the NRC staff finds that the alternative seismic approach appropriately includes the categorization conclusions on seismically-induced failure of relays (i.e., relay chatter) from the Plant A, C, and D case studies.

The NRC staff's review of the licensee's proposed alternative seismic approach, in conjunction with the requirements in 10 CFR 50.69 and the corresponding SoC, finds that the proposed alternative seismic approach provides reasonable confidence in the licensee's evaluations required by 10 CFR 50.69(c)(1)(ii), as well as 10 CFR 50.69(c)(1)(iv) because:

1. The proposed alternative seismic approach includes qualitative consideration of seismic events at several steps of the categorization process, including documentation of the information for presentation to the IDP as part of the integrated, systematic process for categorization.

2. The proposed alternative seismic approach presents system-specific seismic insights to the IDP for consideration as part of the IDP review process as each system is categorized, thereby providing the IDP a means to consider potential impacts of seismic events in the categorization process.

3. The insights presented to the IDP include potentially important seismically-induced failure modes, as well as mitigation capabilities of SSCs during seismically-induced design basis and severe accident events consistent with the conclusions on the determination of unique HSS SSCs from SPRAs in EPRI Report 3002012988. The insights will use plant-specific prior seismic evaluations, and therefore, in conjunction with performance monitoring for the proposed alternative seismic approach, reasonably reflect the current plant configuration. Further, the recommendation for categorizing civil structures in the alternative seismic approach provides appropriate consideration of such failures from a seismic event.

4. The proposed alternative seismic approach presents the IDP with the basis for the proposed alternative seismic approach, including the low seismic hazard for the plant and the criteria for use of the proposed alternative seismic approach.

5. The proposed alternative seismic approach includes qualitative consideration and insights related to the impact of a seismic event on SSCs for each SSC that is categorized and does not limit the scope to SSCs from the case studies supporting this application.

Consideration of Changes to Seismic Hazard at Calvert Cliffs in the Proposed Alternative Seismic
Approach

An important input to the NRC staff's evaluation of the proposed alternative seismic approach is the current knowledge of the seismic hazard at the plant. The possibility exists for the seismic hazard at the site to increase such that the criteria for use of the proposed alternative seismic approach are challenged at Calvert Cliffs. In such a situation, the categorization process may be impacted from a seismic risk perspective either solely due to the seismic risk or by the integrated importance measure determination.

In Attachment 1 of the May 10, 2019, supplement, the licensee stated that "U.S. nuclear power plants that utilize the 50.69 Seismic Alternative (EPRI 3002012988) will continue to compare GMRS to SSE." Since the alternative seismic approach explicitly cites and is based on EPRI Report 3002012988, the continued comparison of GMRS to SSE applies to Calvert Cliffs. The licensee also stated that the seismic hazard at the plant was subject to periodic reconsideration as new information became available through industry evaluations.

Regarding the possibility for the seismic hazard at the site to increase, the licensee provided the following information:

- The criteria for use of the proposed alternative seismic approach provided in the LAR provided a clear and traceable boundary that could be consistently applied to Calvert Cliffs.

- Because the boundary was well defined, if new information was obtained on the site hazard, the continued applicability of the criteria could be confirmed.

- If the Calvert Cliffs seismic hazard changed such that the criteria for use of the proposed alternative seismic approach was no longer valid at Calvert Cliffs, Exelon would follow its categorization review and adjustment process procedures to review the changes to the plant and update, as appropriate, the SSC categorization in accordance with 10 CFR 50.69(e).

- The licensee stated in Attachment 1 of its letter dated May 10, 2019, that if significant changes to the plant risk profile are identified, or if it is identified that a RISC-3 or RISC-4 SSC can or did prevent a safety-significant function from being satisfied, an immediate evaluation and review would be performed prior to the normally scheduled periodic review.

The NRC staff's evaluation of the licensee's information about the consideration of changes to the seismic hazard at Calvert Cliffs as part of the proposed alternative seismic approach determined the following:

- 10 CFR 50.69(e)(1) states:

  > The licensee shall review changes to the plant, operational practices, applicable plant and industry operational experience, and, as appropriate, update the PRA and SSC categorization and treatment processes. The licensee shall perform this review in a timely manner but no longer than once every two refueling outages.

  Changes to the seismic hazard at a site represent changes to the plant, and therefore, would fall under the requirements of 10 CFR 50.69(e)(1). The seismic hazard change potentially impacts the determination of the treatment, which is a consequence of the categorization. Continued use of the alternative seismic approach to determine the treatment would need to be justified because of the exceedance of the criteria for the use of the proposed alternative seismic approach.

- 10 CFR 50.69(e)(3) states, in part:

  > The licensee shall make adjustments as necessary to the categorization or treatment processes so that the categorization process and results are maintained valid.

  Change in the seismic hazard beyond the criteria for use of the proposed alternative seismic approach potentially impacts the validity of the treatment processes and results. Therefore, continued use of the alternative seismic approach to determine the treatment would need to be justified because of the exceedance of the criteria for the use of the proposed alternative seismic approach.

- With respect to RISC-3 SSCs, 10 CFR 50.69(d)(2)(ii) states, in part:

  > For significant conditions adverse to quality, measures must be taken to provide reasonable confidence that the cause of the condition is determined and corrective action taken to preclude repetition.

The change in the seismic hazard beyond the criteria for use of the proposed alternative seismic approach can potentially result in a significant condition adverse to quality. Further, since the condition is the change in the seismic hazard it is highly unlikely to either be corrected or precluded, as required by 10 CFR 50.69(d)(2)(ii), without a change to the categorization approach for the seismic hazard.

- Any change to the alternative seismic approach to account for the impact of the change in the seismic hazard beyond the criteria for use of the proposed alternative seismic approach to meet 10 CFR 50.69(d)(2)(ii) and 10 CFR 50.69(e)(3) would trigger the license condition, which requires prior NRC approval for a change to the proposed alternative seismic approach.

Therefore, based on its review, the NRC staff finds that the proposed alternative seismic approach acceptably includes consideration of changes to the seismic hazard at Calvert Cliffs that exceed the criteria for use of the proposed alternative seismic approach because: (i) the criteria for use of the proposed alternative seismic approach are clear and traceable, (ii) the proposed alternative seismic approach includes periodic reconsideration of the seismic hazard at Calvert Cliffs as new information becomes available, (iii) the proposed alternative seismic approach satisfies the requirements in 10 CFR 50.69 discussed above, and (iv) the licensee has included a proposed license condition in the LAR.

Monitoring of Inputs to and Outcome of Proposed Alternative Seismic Approach

In Section 3.5 of Attachment 1 to the letter dated May 10, 2019, the licensee stated that its configuration control process ensured that changes to the plant, including physical changes and changes to documents, are evaluated to determine the impact on design bases, licensing documents, programs, procedures, and training. The licensee further stated that its performance monitoring process required periodic review to assess changes that could impact the categorization results and to provide the IDP with an opportunity to recommend categorization and treatment adjustments due to such changes. The licensee explained that its configuration control program had been updated to have a checklist that would include:

- A review of the impact on the SCD for configuration changes that may impact a categorized system under 10 CFR 50.69.

- Steps to be performed if redundancy, diversity, or separation requirements are identified or affected, including identification of any potential seismic interaction between added or modified components and new or existing safety-related or safe shutdown components or structures.

- Review of impact to seismic loading and SSE seismic requirements, as well as the method of combining seismic components.

- Review of seismic dynamic qualification of components if the configuration change adds, relocates, or alters seismic Category I mechanical or electrical components.

The licensee stated that its performance monitoring program required that SCDs not be approved by the IDP until the panel's comments on issues, including system-specific seismic insights, had been resolved to the satisfaction of the IDP.

The licensee explained that its scheduled periodic reviews would occur no longer than once every two refueling outages and would evaluate new insights resulting from available risk information (i.e., PRA model or other analysis used in the categorization) changes, design changes, operational changes, and SSC performance. If it was determined that these changes affect the risk information or other elements of the categorization process such that the categorization results are more than minimally affected, then the risk information and the categorization process would be updated. In addition, the licensee stated that if significant changes to the plant risk profile are identified, or if it is identified that a RISC-3 or RISC-4 SSC can or did prevent a safety-significant function from being satisfied, an immediate evaluation and review would be performed prior to the normally scheduled periodic review.

The NRC staff evaluated the licensee's discussion of its performance monitoring program for the proposed alternative seismic approach to ensure (1) the continued validity of the plant-specific information related to the seismic hazard that was developed for each SSC that is categorized, (2) that any changes to the plant, including the seismic hazard, are captured and appropriately addressed as part of the 10 CFR 50.69 program, and (3) that the requirements in 10 CFR 50.69(e) were met for the proposed alternative seismic approach.

The NRC staff finds that the licensee's configuration control program includes consideration of seismic issues and failure modes such as interaction between components and review of seismic loading, as well as seismic dynamic qualification. Further, the performance monitoring program assesses changes that impact the categorization results and provides the IDP with an opportunity to recommend categorization and treatment adjustments due to such changes. Therefore, the licensee's performance monitoring and configuration control process addresses plant-specific seismic consideration, thereby ensuring that the bases for the seismic insights included in SSC categorization continue to remain valid and, if necessary, are presented to the IDP for categorization changes.

The NRC staff also concludes that the licensee's performance monitoring program for 10 CFR 50.69 has the capability to identify significant changes to the plant risk profile, as well as instances of a RISC-3 or RISC-4 SSC not performing a safety-significant function and results in an immediate evaluation and review for such instances. The NRC staff further finds that the impact of changes to the seismic hazard at the plant is included in the proposed alternative seismic approach at Calvert Cliffs. Based on its review and findings, the NRC staff finds that the requirements in 10 CFR 50.69(e) are met for the proposed alternative seismic approach.

The NRC staff's review determined that (1) the licensee's programs provide reasonable assurance that the existing seismic capacity of LSS components would not be significantly impacted, and (2) the monitoring and configuration control program ensures that potential degradation of the seismic capacity would be detected and addressed before significantly impacting the plant risk profile. Therefore, the NRC staff finds that reasonable confidence exists that the potential impact of the seismic hazard on the categorization is maintained acceptably low and the requirements in 10 CFR 50.69(c)(2)(iv) are met for the proposed alternative seismic approach.

3.5.3.2 Other External Hazards (High Winds, External Floods)

The licensee discussed its consideration of non-seismic external hazards and other hazards in Section 3.2.4 of the enclosure to the LAR. Non-seismic external hazards include high winds, external flood hazards, and other hazards listed in Appendix 6-A of the 2009 ASME/ANS PRA Standard (RA-Sa-2009). The licensee evaluated all non-seismic external hazards and other

hazards for the 10 CFR 50.69 application using a plant-specific evaluation in accordance with Generic Letter 88-20 and the criteria in the 2009 ASME/ANS PRA Standard. The NRC staff reviewed the licensee's evaluation, which was provided in Attachment 4 of the enclosure to the LAR.

The NRC staff assessment for the licensee's focused evaluation for its reevaluated external flood hazard (Reference 49) explains that the current design basis for the licensee only addresses local intense precipitation (LIP) at certain buildings (EDG and station blackout buildings), and, at those buildings, the amended (i.e., revised) reevaluated LIP hazard is bound. The licensee's analysis of other locations, including the south service building, turbine building, auxiliary building, and the diesel generator building) determined that LIP elevations at these locations were higher than that mentioned in the current design basis but lower than the plant grade elevation (or finished floor elevation at door entrance). Furthermore, the NRC staff assessment for focused evaluation found that the licensee relied on a passive feature, the plant grade, to justify that there is available physical margin for LIP at the locations not in the current design basis; that the licensee did not consider ground infiltration and assumed that the site drainage system was non-functional; and that the door openings to structures containing SSCs have a curb that was not included in the determination of available physical margin. In the letter dated July 1, 2019, the licensee stated that SSCs credited for screening of external hazards will be evaluated according to the flow chart in NEI 00-04, Figure 5-6. Based on the review of the information provided by the licensee in Attachment 4 of the enclosure to the LAR, the licensee's letter dated July 1, 2019, as well as the NRC staff's previous assessment for the licensee's focused evaluation for the reevaluated external flood hazard, the NRC staff finds that (1) the licensee's design basis is a bounding or demonstrably conservative analysis for all external flood hazards, (2) conservatisms exist in the evaluation performed for available physical margin in Calvert Cliffs focused evaluation for its reevaluated flood hazard, and (2) the licensee's SSC categorization process will evaluate the safety significance of any SSCs for the external flooding hazard consistent with the guidance provided in NEI 00-04, as endorsed by the NRC.

Section 5.A.3.1.9 of the licensee's Updated Final Safety Analysis Report (UFSAR; Reference 50) provides the design basis for tornado hazard. The NRC staff's review of the design basis in Section 5.A.3.1.9 of the UFSAR and NUREG/CR-4461, "Tornado Climatology of the Contiguous United States" (Reference 51), finds that the occurrence frequency of the design-basis wind speeds for Calvert Cliffs is less $1x10^{-6}$ per year. The NRC staff's review also notes that the primary concern for high straight winds is loss-of-offsite power caused by the winds and that the internal events PRA already includes loss-of-offsite power events due to severe weather, including high, straight winds. Therefore, the NRC staff finds that the impact of high straight winds on plant response and the resulting categorization of SSCs is included in the categorization process. Section 5.3.1 of the licensee's UFSAR states that tornado-generated missiles neither penetrate the containment structure wall nor endanger the structural integrity of the containment structure or any components of the reactor coolant system. Section 5.A.3.1.9 of the licensee's UFSAR includes a discussion of a probabilistic evaluation for determining the need to provide missile protection for several SSCs. The licensee's evaluation, as discussed in the UFSAR, concluded that such protection was not required based on the aggregate probability of exposure of those SSCs to tornado-generated missiles.

The discussion provided for extreme winds or tornados in Attachment 4 of the enclosure to the LAR states that NEI 00-04 requires that, as part of the external hazard screening, an evaluation be conducted to determine if there are components that participate in screened scenarios and whose failure would result in an unscreened scenario and that such SSCs are required to be high safety-significant in the categorization process. Based on its review, the NRC staff finds

that the licensee's SSC categorization process will evaluate the safety significance of SSCs for the extreme winds or tornado hazard consistent with the guidance provided in NEI 00-04, as endorsed by the NRC.

In the licensee's letter dated July 1, 2019, the licensee stated that SSCs credited for screening of external hazards will be evaluated according to the flow chart in NEI 00-04, Figure 5-6. Based on its review, the NRC finds that the licensee's SSC categorization process will evaluate the safety significance of SSCs for non-seismic external hazards and other hazards in Attachment 4 of the enclosure to the LAR consistent with the guidance provided in Figure 5-6 of NEI 00-04, Revision 0, as endorsed by the NRC in RG 1.201, Revision 1.

3.5.3.3 Shutdown Risk

Section 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 guidance in NEI 00-04, the licensee proposed using the shutdown safety assessment based on NUMARC 91-06 (Reference 26). 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. NUMARC 91-06 specifies that a DID approach should be used with respect to each defined shutdown key safety function. This is accomplished by designating a running and an alternative system/train to accomplish the given key safety function.

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

3.5.4 Component Safety Significance Assessment for Passive Components

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

In Section 3.1.2 of the LAR, the licensee proposed using a categorization method for passive components not cited in NEI 00-04 or RG 1.201, Revision 1, for passive component categorization but which was approved by the NRC for ANO-2 (Reference 16). The ANO-2 methodology is a risk-informed safety classification and treatment program for repair/replacement activities for Class 2 and Class 3 pressure-retaining items and their associated supports (exclusive of Class CC and MC items), using a modification of the ASME Code Case N-660, "Risk-Informed Safety Classification for Use in Risk-Informed Repair/Replacement Activities, Section XI, Division 1" (Reference 52). 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.

The licensee stated in the LAR that all ASME Code Class 1 SSCs with a pressure-retaining function, as well as supports, will be assigned HSS for passive categorization that 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 staff finds the licensee's proposed approach for passive categorization is acceptable for the 10 CFR 50.69 categorization process.

3.5.5 Summary

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

- PRA to assess internal events, including internal flooding risk
- FPRA to assess fire risk
- EPRI alternative approach to assess seismic risk
- Screening using IPEEE to assess risk from other external hazards (high winds, external floods)
- Shutdown safety assessment process to assess shutdown risk
- ANO-2 (see Reference 11) passive categorization method to assess passive component risk for Class 2, Class 3, and non-Code class SSCs and their associated supports

3.6 DID (NEI 00-04, Section 6)

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

Figure 6-1 in NEI 00-04 provides guidance to assess design-basis DID based on the frequency of the design-basis internal initiating event and the number of redundant and diverse trains nominally available to mitigate the initiating event. For each initiating event frequency, components are assigned as HSS if fewer than the indicated number of mitigating trains are nominally available. Section 6 of NEI 00-04 also provides guidance to assess containment DID based on preserving containment isolation and long-term containment integrity and on preventing containment bypass and early hydrogen burns.

RG 1.201, Revision 1, endorses the guidance in Section 6 of NEI 00-04 but notes that the containment isolation criteria in this section of the guidance are separate and distinct from those set forth in 10 CFR 50.69(b)(1)(x). The criteria in 10 CFR 50.69(b)(1)(x) are to be used in determining which containment penetrations and valves may be exempted from the Type B and

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

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

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

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

In LAR Section 3.1.1, the licensee stated that if any SSC is identified as HSS from either the integrated risk component safety significance assessment (Section 5 of NEI 00 04), the DID assessment (Section 6 of NEI 00-04), or the qualitative criteria (Section 9 of NEI 00 04), the associated system function(s) would be identified as HSS. The licensee also stated that once a system function is identified as HSS, then all the components that support that function are preliminary HSS. Table 3-1 of the LAR explains that 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., internal events PRA and integrated PRA importance measures described in Section 5.6 of NEI 00-04). LAR Section 3.1.1 further states that components that are identified as HSS from using the non-PRA approaches, except the alternative seismic approach (shutdown risk, and other external hazards) will remain HSS (IDP cannot override) regardless of the significance of the functions to which they are mapped.

The NRC staff finds that the above description provided by the licensee for the preliminary categorization process is consistent with the guidance in NEI 00-04, as endorsed in RG 1.201, Revision 1, and is, therefore, acceptable.

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

Section 50.69(c)(1)(iv) of 10 CFR requires, in part, that any potential increases in CDF and LERF resulting from changes to treatment are small. The categorization process described in Section 8 of NEI 00-04, as endorsed by RG 1.201, Revision 1, includes an overall risk sensitivity study for all the LSS components to assure that if the unreliability of the components is increased, the increase in risk would be small (i.e., meet the acceptance guidelines of RG 1.174, Revision 3). Sections 3.1.1 and 3.2.7 of the LAR clarify that in the sensitivity study, the unreliability of all LSS SSCs modeled in the PRA(s) will be increased by a factor of 3. Separate sensitivity studies are to be performed for each system categorized, as well as a cumulative sensitivity study for all the SSCs categorized through the 10 CFR 50.69 process.

This sensitivity study, together with the periodic review process, assure that the potential cumulative risk increase from the categorization is maintained acceptably low. The performance monitoring process monitors the component performance to ensure that potential increases in failure rates of categorized components are detected and addressed before reaching the rate assumed in the sensitivity study. The NRC staff finds that the licensee will

perform the risk sensitivity study consistent with the guidance in Section 8 of NEI 00-04, and therefore, will assure that the potential cumulative risk increase from the categorization is maintained acceptably low, as required by 10 CFR 50.69(c)(1)(iv).

3.9 IDP Review and SSC Categorization (NEI 00-04, Sections 9 and 10)

As required by 10 CFR 50.69(c)(2), the SSCs must be categorized by an IDP staffed with expert plant-knowledgeable members whose expertise includes, at a minimum, PRA, safety analysis, plant operations, design engineering, and system engineering. 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 IDP will include the required expertise.

The guidance in NEI 00-04, as endorsed in RG 1.201, Revision 1, provides confidence that the IDP expertise is sufficient to perform the categorization and that the results of the different evaluations (PRA and non-PRA) are used in an integrated, systematic process, as required by 10 CFR 50.69(c)(1)(ii).

In Section 3.1.1 of the LAR, the licensee stated 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. The licensee further stated that the IDP will be trained in the specific technical aspects and requirements related to the categorization process. This 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, Revision 1. Therefore, all aspects of the integrated, systematic process used to characterize SSCs will reasonably reflect current plant configuration and operating practices and applicable plant and industry operational experience as required by 10 CFR 50.69(c)(1)(ii).

The licensee stated in Section 3.1.1 of the LAR that the assessments of the qualitative considerations are agreed upon by the IDP in accordance with Section 9.2 of NEI 00-04. 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. These 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 or not confirming 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 final assessment of the qualitative criteria is the direct responsibility of the IDP. If the IDP determines that any one of the qualitative criteria 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, is acceptable.

The IDP may change the categorization of a component from LSS to HSS based on its assessment and decisionmaking. As outlined in NEI 00-04, Section 10.2, and confirmed in the LAR, the IDP may recategorize 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, FPRA, integrated PRA component risk, alternative EPRI seismic approach, 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.

Section 50.69(c)(1)(iv) of 10 CFR requires, in part, reasonable confidence that sufficient safety margins are maintained for SSCs categorized as RISC-3. The licensee addresses safety margins through an integrated engineering evaluation that would nominally be addressed by the IDP. Consistent with the discussion in NEI 00-04 guidance endorsed by RG 1.201, Revision 1, the IDP need not explicitly consider safety margins. Sufficient safety margins 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 increase 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)

Section 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(f) and 10 CFR 50.69(e), respectively. Maintaining change control and periodic review will also maintain confidence that all aspects of the program reflect current plant operation.

Section 50.69(e) of 10 CFR requires periodic updates to the licensee's PRA and SSC categorization. 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 described in LAR Section 3.2.6, the licensee has administrative controls in place to ensure that the PRA models used to support the categorization reflect the as-built, as-operated plant over time. The licensee's process includes regularly scheduled and interim (as needed) PRA model updates. The process includes provisions for monitoring issues affecting the PRA models (e.g., due to changes in the plant, errors or limitations identified in the model, industry operational experience) for assessing the risk impact of unincorporated changes and for

controlling the model and associated computer files. The process also includes reevaluating previously categorized systems to ensure the continued validity of the categorization. Routine PRA updates are performed every two refueling cycles at a minimum.

The NRC staff finds that this description is consistent with the requirements for feedback and process adjustments required by 10 CFR 50.69(e), and is, therefore, acceptable.

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 including the following ten elements:

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

In addition, LAR Attachment 1 (List of Categorization Prerequisites) states that the licensee will establish procedures for the use of the categorization process that 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, as endorsed by RG 1.201, Revision 1, and are in conformance with the requirements of 10 CFR 50.69(f)(1). Therefore, the NRC staff finds the documentation and records acceptable.

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

3.11 10 CFR 50.69 Implementation License Condition

Section 50.69(b)(2) of 10 CFR requires, in part, 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 a licensee's implementation of this section if it determines that the categorization process satisfies the requirements of 10 CFR 50.69(c). As described in this SE, the NRC staff has concluded that the 10 CFR 50.69 categorization process described in the licensee's application, as supplemented, includes a description of the categorization process that satisfies the requirements of 10 CFR 50.69(c).

In the LAR, as supplement by letter dated October 30, 2019, the licensee proposed to amend its renewed facility operating licenses by adding the following license condition that would allow for the implementation of 10 CFR 50.69:

> Exelon is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 Structures, Systems, and Components (SSCs) using: Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events, including internal flooding, and internal fire; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 and non-Class SSCs and their associated supports; the results of the non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009 for other external hazards except seismic; and the alternative seismic approach as described in Exelon's original submittal letter dated November 28, 2018, and all its subsequent associated supplements as specified in License Amendment No. [XXX] dated [DATE].
>
> 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 review of the licensee's LAR and the evaluation in this SE, the NRC staff finds that the proposed license condition is acceptable because it adequately implements 10 CFR 50.69 using models, methods, and approaches that are acceptable to the NRC and is consistent with applicable guidance that has previously been endorsed as acceptable by the NRC.

3.12 Technical Conclusion

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

(1) Considers results and insights from plant-specific internal events (including internal flooding), and FPRAs, which are of sufficient quality and level of detail to

support the categorization process and that have been subjected to a peer review process against RG 1.200, Revision 2, as reviewed in Section 3.5.1 of this SE, and therefore, meets the requirements in 10 CFR 50.69(c)(1)(i).

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

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

(4) Includes evaluations that provide reasonable confidence that for SSCs categorized as RISC-3, sufficient safety margins are maintained and that any potential increases in CDF and LERF resulting from changes in treatment are small, as reviewed in 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.

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

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Maryland State official was notified of the proposed issuance of the amendments on November 14, 2019. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding published in the *Federal Register* (84 FR 494; January 30, 2019). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

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

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Date: February 28, 2020

SUBJECT: CALVERT CLIFFS NUCLEAR POWER PLANT, UNITS 1 AND 2 – ISSUANCE OF AMENDMENT NOS. 332 AND 310 RE: RISK-INFORMED CATEGORIZATION AND TREATMENT OF STRUCTURES, SYSTEMS, AND COMPONENTS FOR NUCLEAR POWER REACTORS (EPID L-2018-LLA-0482) DATED FEBRUARY 28, 2020

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