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

October 22, 2018

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

SUBJECT: BRAIDWOOD 1 & 2, BYRON 1 & 2 - ISSUANCE OF AMENDMENTS NOS 198, 198, 204, AND 204, RESPECTIVELY, REGARDING ADOPTION OF TITLE 10 OF THE CODE OF FEDERAL REGULATIONS SECTION 50.69, "RISK-INFORMED CATEGORIZATION AND TREATMENT OF STRUCTURES, SYSTEMS AND COMPONENTS FOR NUCLEAR POWER REACTORS" (CAC NOS. MG0201, MG0202, MG0203, AND MG204; EPID L-2017-LLA-0285)

Dear Mr. Hanson:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 198 to Renewed Facility Operating License No. NPF-72 and Amendment No. 198 to Renewed Facility Operating License No. NPF-77 for the Braidwood Station, Units 1 and 2, respectively, and Amendment No. 204 to Renewed Facility Operating License No. NPF-37 and Amendment No. 204 to Renewed Facility Operating License No. NPF-66 for the Byron Station, Unit Nos. 1 and 2, respectively. The amendments are in response to your application dated September 1, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17244A093), as supplemented by letters dated April 4, 2018 (ADAMS Accession No. ML18094A955), June 13, 2018 (ADAMS Accession No. ML18165A181), and September 13, 2018 (ADAMS Accession No. ML18256A392).

The amendments added a new license condition to the Renewed Facility Operating Licenses to allow the implementation of risk-informed categorization and treatment of structures, systems, and components for nuclear power reactors in accordance with Title 10 of the *Code of Federal Regulations* Section 50.69.

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

Sincerely,

Joel S. Wiebe

Joel S. Wiebe, Senior Project Manager
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. STN 50-456, STN 50-457,
STN 50-454, and STN 50-455

Enclosures:
1. Amendment No. 198 to NPF-72
2. Amendment No. 198 to NPF-77
3. Amendment No. 204 to NPF-37
4. Amendment No. 204 to NPF-66
5. Safety Evaluation

cc: Listserv

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

EXELON GENERATION COMPANY, LLC

DOCKET NO. STN 50-456

BRAIDWOOD STATION, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 198
Renewed License No. NPF-72

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

   A. The application for amendment by Exelon Generation Company, LLC (the licensee) dated September 1, 2017, as supplemented by letters dated April 4, 2018, June 13, 2018, and September 13, 2018, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Renewed Facility Operating License as indicated in the attachment to this license amendment.

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

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment: Changes to the Renewed Facility Operating License

Date of Issuance: October 22, 2018

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

EXELON GENERATION COMPANY, LLC

DOCKET NO. STN 50-457

BRAIDWOOD STATION, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 198
Renewed License No. NPF-77

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

   A. The application for amendment by Exelon Generation Company, LLC (the licensee) dated September 1, 2017, as supplemented by letters dated April 4, 2018, June 13, 2018, and September 13, 2018, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Renewed Facility Operating License as indicated in the attachment to this license amendment.

Enclosure 2

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

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment: Changes to the Renewed Facility Operating License

Date of Issuance: October 22, 2018

ATTACHMENT TO LICENSE AMENDMENT NOS. 198 AND 198

RENEWED FACILITY OPERATING LICENSE NOS. NPF-72 AND NPF-77

BRAIDWOOD STATION, UNITS 1 AND 2

DOCKET NOS. STN 50-456 AND STN 50-457

Replace the following pages of the Renewed Facility Operating Licenses with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
License NPF-72	License NPF-72
Page 4	Page 4 (pagination)
Page 5	Page 5 (pagination)
Page 6	Page 6 (pagination)
Page 7	Page 7
License NPF-77	License NPF-77
Page 4	Page 4 (pagination)
Page 5	Page 5 (pagination)
Page 6	Page 6 (pagination)
Page 7	Page 7
Page 8	page 8 (pagination)

(3) Emergency Planning

In the event that the NRC finds that the lack of progress in completion of the procedures in the Federal Emergency Management Agency's final rule, 44 CFR Part 350, is an indication that a major substantive problem exists in achieving or maintaining an adequate state of emergency preparedness, the provision of 10 CFR Section 50.54(s)(2) will apply.

(4) Deleted.

(5) Deleted.

(6) Deleted.

(7) Additional Conditions

The Additional Conditions contained in Appendix C, as revised through Amendment No. 193, are hereby incorporated into this renewed license. The licensee shall operate the facility in accordance with the Additional Conditions.

(8) Exelon Generation Company shall provide to the Director of the Office of Nuclear Reactor Regulation a copy of any application, at the time it is filed, to transfer (excluding grants of security interests or liens) from Exelon Generation Company to its direct or indirect parent, or to any other affiliated company, facilities for the production, transmission, or distribution of electric energy having a depreciated book value exceeding ten percent (10%) of Exelon Generation Company's consolidated net utility plant, as recorded on Exelon Generation Company's books of account.

(9) Exelon Generation Company shall have decommissioning trust funds for Braidwood, Unit 1, in the following minimum amount, when Braidwood, Unit 1, is transferred to Exelon Generation Company:

Braidwood Unit 1	$154,273,345

(10) The decommissioning trust agreement for Braidwood, Unit 1, at the time the transfer of the unit to Exelon Generation Company is effected and thereafter, is subject to the following:

(a) The decommissioning trust agreement must be in a form acceptable to the NRC.

(b) With respect to the decommissioning trust fund, investments in the securities or other obligations of Exelon Corporation or affiliates thereof, or their successors or assigns are prohibited. Except for investments tied to market indexes or other non-nuclear sector mutual funds, investments in any entity owning one or more nuclear power plants are prohibited.

(c) The decommissioning trust agreement for Braidwood, Unit 1, must provide that no disbursements or payments from the trust shall be made by the trustee unless the trustee has first given the Director of the Office of Nuclear Reactor Regulation 30 days prior written notice of payment. The decommissioning trust agreement shall further contain a provision that no disbursements or payments from the trust shall be made if the trustee receives prior written notice of objection from the NRC.

(d) The decommissioning trust agreement must provide that the agreement can not be amended in any material respect without 30 days prior written notification to the Director of the Office of Nuclear Reactor Regulation.

(e) The appropriate section of the decommissioning trust agreement shall state that the trustee, investment advisor, or anyone else directing the investments made in the trust shall adhere to a "prudent investor" standard, as specified in 18 CFR 35.32(a)(3) of the Federal Energy Regulatory Commission's regulations.

(11) Exelon Generation Company shall take all necessary steps to ensure that the decommissioning trust is maintained in accordance with the application for approval of the transfer of the Braidwood, Unit 1, license and the requirements of the Order approving the transfer, and consistent with the safety evaluation supporting the Order.

(12) Mitigation Strategy License Condition

Develop and maintain strategies for addressing large fires and explosions and 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

(13) <u>License Renewal License Conditions</u>

(a) The information in the UFSAR supplement, submitted pursuant to 10 CFR 54.21(d), as revised during the license renewal application review process, and as supplemented by the Commitments applicable to Braidwood Unit 1 in Appendix A of the "Safety Evaluation Report Related to the License Renewal of Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2" (SER) dated July 2015, is collectively the "License Renewal UFSAR Supplement." This Supplement is henceforth part of the UFSAR which will be updated in accordance with 10 CFR 50.71(e). As such, the licensee may make changes to the programs and activities applicable to Braidwood Unit 1 described in this Supplement provided the licensee evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

(b) This License Renewal UFSAR Supplement, as revised per License Condition 13(a) above, describes certain programs to be implemented and activities to be completed prior to the period of extended operation.

1. The licensee shall implement those new programs and enhancements to existing programs no later than April 17, 2026.

2. The licensee shall complete those activities as noted in the Commitments applicable to Braidwood Unit 1 in this Supplement no later than April 17, 2026 or the end of the last refueling outage prior to the period of extended operation, whichever occurs later.

3. The licensee shall notify the NRC in writing within 30 days after having accomplished item (b)1 above and include the status of those activities that have been or remain to be completed in item (b)2 above.

(c) The flux thimble tube corrective actions, inspections, and replacements identified in the SER, Commitment No. 24, for Braidwood Units 1 and 2, shall be implemented in accordance with the schedule in the Commitment. Periodic eddy current testing/inspections of all flux thimble tubes shall be performed at least every two refueling outages, and the data shall be trended and retained in auditable form. A flux thimble tube shall not remain in service for more than two (2) operating fuel cycles without successful completion of eddy current testing for that thimble tube.

(14) Adoption of 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems, and components for nuclear power plants"

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, Class 3, and non-Code class SSCs and their associated supports; and the results of non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards, i.e., seismic margin analysis (SMA) to evaluate seismic risk, and a screening of other external hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009; as specified in the license amendment No. 198, dated October 22, 2018.

Exelon will complete the updated implementation items listed in Attachment 1 of Exelon letter to NRC dated September 13, 2018, prior to implementation of 10 CFR 50.69. All issues identified in the attachment will be addressed and any associated changes will be made, focused scope peer reviews will be performed on changes that are PRA upgrades as defined in the PRA standard (ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, Revision 2), and any findings will be resolved and reflected in the PRA of record prior to implementation of the 10 CFR 50.69 categorization process.

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

D. An exemption was previously granted pursuant to 10 CFR 70.24. The exemption was granted with NRC materials license No. SNM-1938, issued October 8, 1985, and relieved the licensee from the requirement of having a criticality alarm system. Therefore, the licensee is exempted from the criticality alarm system provision of 10 CFR 70.24 so far as this section applies to the storage of fuel assemblies held under this license.

E. The licensee shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report, as supplemented and amended, and as approved in the SER dated November 1983 and its supplements, subject to the following provision:

The licensee may make changes to the approved fire protection program without prior approval of the Commission, only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

(3) Emergency Planning

In the event that the NRC finds that the lack of progress in completion of the procedures in the Federal Emergency Management Agency's final rule, 44 CFR Part 350, is an indication that a major substantive problem exists in achieving or maintaining an adequate state of emergency preparedness, the provision of 10 CFR Section 50.54(s)(2) will apply.

(4) Deleted.

(5) Deleted.

(6) Additional Conditions

The Additional Conditions contained in Appendix C, as revised through Amendment No. 193, are hereby incorporated into this renewed license. The licensee shall operate the facility in accordance with the Additional Conditions.

(7) Exelon Generation Company, LLC, shall provide to the Director of the Office of Nuclear Reactor Regulation a copy of any application, at the time it is filed, to transfer (excluding grants of security interests or liens) from Exelon Generation Company, LLC, to its direct or indirect parent, or to any other affiliated company, facilities for the production, transmission, or distribution of electric energy having a depreciated book value exceeding ten percent (10%) of Exelon Generation Company's consolidated net utility plant, as recorded on Exelon Generation Company, LLC's books of account.

(8) Exelon Generation Company, LLC, shall have decommissioning trust funds for Braidwood, Unit 2, in the following minimum amount, when Braidwood, Unit 2, is transferred to Exelon Generation Company, LLC:

Braidwood Unit 2	$154,448,967

(9) The decommissioning trust agreement for Braidwood, Unit 2, at the time the transfer of the unit to Exelon Generation Company, LLC is effected and thereafter, is subject to the following:

(a) The decommissioning trust agreement must be in a form acceptable to the NRC.

(b) With respect to the decommissioning trust fund, investments in the securities or other obligations of Exelon Corporation or affiliates thereof, or their successors or assigns are prohibited. Except for investments tied to market indexes or other non-nuclear sector mutual funds, investments in any entity owning one or more nuclear power plants are prohibited.

(c) The decommissioning trust agreement for Braidwood, Unit 2, must provide that no disbursements or payments from the trust shall be made by the trustee unless the trustee has first given the Director of the Office of Nuclear Reactor Regulation 30 days prior written notice of payment. The decommissioning trust agreement shall further contain a provision that no disbursements or payments from the trust shall be made if the trustee receives prior written notice of objection from the NRC.

(d) The decommissioning trust agreement must provide that the agreement can not be amended in any material respect without 30 days prior written notification to the Director of the Office of Nuclear Reactor Regulation.

(e) The appropriate section of the decommissioning trust agreement shall state that the trustee, investment advisor, or anyone else directing the investments made in the trust shall adhere to a "prudent investor" standard, as specified in 18 CFR 35.32(a)(3) of the Federal Energy Regulatory Commission's regulations.

(10) Exelon Generation Company, LLC shall take all necessary steps to ensure that the decommissioning trust is maintained in accordance with the application for approval of the transfer of the Braidwood, Unit 2, license and the requirements of the Order approving the transfer, and consistent with the safety evaluation supporting the Order.

(11) <u>Mitigation Strategy License Condition</u>

Develop and maintain strategies for addressing large fires and explosions and 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

(12) License Renewal License Conditions

(a) The information in the UFSAR supplement, submitted pursuant to 10 CFR 54.21(d), as revised during the license renewal application review process, and as supplemented by the Commitments applicable to Braidwood Unit 2 in Appendix A of the "Safety Evaluation Report Related to the License Renewal of Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2" (SER) dated July 2015, is collectively the "License Renewal UFSAR Supplement." This Supplement is henceforth part of the UFSAR which will be updated in accordance with 10 CFR 50.71(e). As such, the licensee may make changes to the programs and activities applicable to Braidwood Unit 2 described in this Supplement provided the licensee evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

(b) This License Renewal UFSAR Supplement, as revised per License Condition 12(a) above, describes certain programs to be implemented and activities to be completed prior to the period of extended operation.

1. The licensee shall implement those new programs and enhancements to existing programs no later than June 18, 2027.

2. The licensee shall complete those activities as noted in the Commitments applicable to Braidwood Unit 2 in this Supplement no later than June 18, 2027 or the end of the last refueling outage prior to the period of extended operation, whichever occurs later.

3. The licensee shall notify the NRC in writing within 30 days after having accomplished item (b)1 above and include the status of those activities that have been or remain to be completed in item (b)2 above.

(c) The flux thimble tube corrective actions, inspections, and replacements identified in the SER, Commitment No. 24, for Braidwood Units 1 and 2, shall be implemented in accordance with the schedule in the Commitment. Periodic eddy current testing/inspections of all flux thimble tubes shall be performed at least every two refueling outages, and the data shall be trended and retained in auditable form. A flux thimble tube shall not remain in service for more than two (2) operating fuel cycles without successful completion of eddy current testing for that thimble tube.

(d) The Braidwood Unit 2 reactor head closure stud hole location No. 35 will be repaired no later than June 18, 2027, or before the end of the last refueling outage prior to the period of extended operation (whichever occurs later), so that all 54 reactor head closure studs are operable and tensioned during the period of extended operation.

(13) Adoption of 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems, and components for nuclear power plants"

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, Class 3, and non-Code class SSCs and their associated supports; and the results of non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards, i.e., seismic margin analysis (SMA) to evaluate seismic risk, and a screening of other external hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009; as specified in the license amendment No. 198, dated October 22, 2018.

Exelon will complete the updated implementation items listed in Attachment 1 of Exelon letter to NRC dated September 13, 2018, prior to implementation of 10 CFR 50.69. All issues identified in the attachment will be addressed and any associated changes will be made, focused scope peer reviews will be performed on changes that are PRA upgrades as defined in the PRA standard (ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, Revision 2), and any findings will be resolved and reflected in the PRA of record prior to implementation of the 10 CFR 50.69 categorization process.

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

D. An exemption was previously granted pursuant to 10 CFR 70.24. The exemption was granted with NRC materials license No. SNM-1938, issued October 8, 1985, and relieved the licensee from the requirement of having a criticality alarm system. Therefore, the licensee is exempted from the criticality alarm system provision of 10 CFR 70.24 so far as this section applies to the storage of fuel assemblies held under this license.

E. The licensee shall implement and maintain in affect all provisions of the approved fire protection program as described in the Final Safety Analysis Report, as supplemented and amended, and as approved in the SER dated November 1983 and its supplements, subject to the following provision:

The licensee may make changes to the approved fire protection program without prior approval of the Commission, only if these changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

F. Exelon Generation Company shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualifications, 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[1], which contain Safeguards Information protected under 10 CFR 73.21, is entitled: "Braidwood Station Security Plan, Training and Qualification Plan, and Safeguards Contingency Plan, Revision 3," submitted by letter dated May 17, 2006.

Exelon Generation Company 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 Exelon Generation Company CSP was approved by License Amendment No.168 and modified by License Amendment No. 185.

[1] The training and Qualification Plan and Safeguards Contingency Plan are Appendices to the Security Plan.

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

EXELON GENERATION COMPANY, LLC

DOCKET NO. STN 50-454

BYRON STATION, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 204
Renewed License No. NPF-37

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

   A. The application for amendment by Exelon Generation Company, LLC (the licensee) dated September 1, 2017, as supplemented by letters dated April 4, 2018, June 13, 2018, and September 13, 2018, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Renewed Facility Operating License as indicated in the attachment to this license amendment.

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

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment: Changes to the Renewed Facility Operating License

Date of Issuance: October 22, 2018

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

EXELON GENERATION COMPANY, LLC

DOCKET NO. STN 50-455

BYRON STATION, UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 204
Renewed License No. NPF-66

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

   A. The application for amendment by Exelon Generation Company, LLC (the licensee) dated September 1, 2017, as supplemented by letters dated April 4, 2018, June 13, 2018, and September 13, 2018, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Renewed Facility Operating License as indicated in the attachment to this license amendment.

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

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment: Changes to the Renewed Facility Operating License

Date of Issuance: October 22, 2018

ATTACHMENT TO LICENSE AMENDMENT NOS. 204 AND 204

RENEWED FACILITY OPERATING LICENSE NOS. NPF-37 AND NPF-66

BYRON STATION, UNIT NOS. 1 AND 2

DOCKET NOS. STN 50-454 AND STN 50-455

Replace the following pages of the Renewed Facility Operating Licenses with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
License NPF-37	License NPF-37
Page 7	Page 7
Page 8	Page 8
License NPF-66	License NPF-66
Page 4	Page 4 (pagination)
Page 5	Page 5 (pagination)
Page 6	Page 6
Page 7	Page 7

(23) License Renewal License Conditions

(a) The information in the UFSAR supplement, submitted pursuant to 10 CFR 54.21(d), as revised during the license renewal application review process, and as supplemented by the Commitments applicable to Byron Unit 1 in Appendix A of the "Safety Evaluation Report Related to the License Renewal of Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2" (SER) dated July 2015, is collectively the "License Renewal UFSAR Supplement." This Supplement is henceforth part of the UFSAR which will be updated in accordance with 10 CFR 50.71(e). As such, the licensee may make changes to the programs and activities applicable to Byron Unit 1 described in this Supplement provided the licensee evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

(b) This License Renewal UFSAR Supplement, as revised per License Condition 23(a) above, describes certain programs to be implemented and activities to be completed prior to the period of extended operation.

1. The licensee shall implement those new programs and enhancements to existing programs no later than April 30, 2024.

2. The licensee shall complete those activities as noted in the Commitments applicable to Byron Unit 1 in this Supplement no later than April 30, 2024 or the end of the last refueling outage prior to the period of extended operation, whichever occurs later.

3. The licensee shall notify the NRC in writing within 30 days after having accomplished item (b)1 above and include the status of those activities that have been or remain to be completed in item (b)2 above.

(24) Adoption of 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems, and components for nuclear power plants"

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, Class 3, and non-Code class SSCs and their associated supports; and the results of non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards, i.e., seismic margin analysis (SMA) to evaluate seismic risk, and a screening of other external hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009; as specified in the license amendment No. 204, dated October 22, 2018.

Exelon will complete the updated implementation items listed in Attachment 1 of Exelon letter to NRC dated September 13, 2018, prior to implementation of 10 CFR 50.69. All issues identified in the attachment will be addressed and any associated changes will be made, focused scope peer reviews will be performed on changes that are PRA upgrades as defined in the PRA standard (ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, Revision 2), and any findings will be resolved and reflected in the PRA of record prior to implementation of the 10 CFR 50.69 categorization process.

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

D. The facility requires no exemptions from the requirements of 10 CFR Part 50.

E. Exelon Generation Company shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualifications, 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[1], which contain Safeguards Information protected under 10 CFR 73.21, is entitled: "Byron Nuclear Power Station Security Plan, Training and Qualification Plan, and Safeguards Contingency Plan, Revision 3," submitted by letter dated May 17, 2006.

Exelon Generation Company 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 Exelon Generation Company CSP was approved by License Amendment No.175 and modified by License Amendment No. 191.

[1] The training and Qualification Plan and Safeguards Contingency Plan are Appendices to the Security Plan

(3) Deleted.

(4) Deleted.

(5) Deleted.

(6) <u>Additional Conditions</u>

The Additional Conditions contained in Appendix C, as revised through Amendment No. 198, are hereby incorporated into this renewed license. The licensee shall operate the facility in accordance with the Additional Conditions.

(7) Exelon Generation Company, LLC, shall provide to the Director of the Office of Nuclear Reactor Regulation a copy of any application, at the time it is filed, to transfer (excluding grants of security interests or liens) from Exelon Generation Company, LLC, to its direct or indirect parent, or to any other affiliated company, facilities for the production, transmission, or distribution of electric energy having a depreciated book value exceeding ten percent (10%) of Exelon Generation Company's consolidated net utility plant, as recorded on Exelon Generation Company, LLC's books of account.

(8) Exelon Generation Company, LLC, shall have decommissioning trust funds for Byron, Unit 2, in the following minimum amount, when Byron, Unit 2, is transferred to Exelon Generation Company, LLC:

Byron Unit 2	$156,560,489

(9) The decommissioning trust agreement for Byron, Unit 2, at the time the transfer of the unit to Exelon Generation Company, LLC is effected and thereafter, is subject to the following:

(a) The decommissioning trust agreement must be in a form acceptable to the NRC.

(b) With respect to the decommissioning trust fund, investments in the securities or other obligations of Exelon Corporation or affiliates thereof, or their successors or assigns are prohibited. Except for investments tied to market indexes or other non-nuclear sector mutual funds, investments in any entity owning one or more nuclear power plants are prohibited.

(c) The decommissioning trust agreement for Byron, Unit 2 must provide that no disbursements or payments from the trust shall be made by the trustee unless the trustee has first given the Director of the Office of Nuclear Reactor Regulation, 30 days prior written notice of payment. The decommissioning trust agreement shall further contain a provision that no disbursements or payments from the trust shall be made if the trustee receives prior written notice of objection from the NRC.

(d) The decommissioning trust agreement must provide that the agreement can not be amended in any material respect without 30 days prior written notification to the Director of the Office of Nuclear Reactor Regulation.

(e) The appropriate section of the decommissioning trust agreement shall state that the trustee, investment advisor, or anyone else directing the investments made in the trust shall adhere to a "prudent investor" standard, as specified in 18 CFR 35.32(a)(3) of the Federal Energy Regulatory Commission's regulations.

(10) Exelon Generation Company, LLC shall take all necessary steps to ensure that the decommissioning trust is maintained in accordance with the application for approval of the transfer of the Byron, Unit 2, license and the requirements of the Order approving the transfer, and consistent with the safety evaluation supporting the Order.

(11) Mitigation Strategy License Condition

Develop and maintain strategies for addressing large fires and explosions and 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

(12) License Renewal License Conditions

(a) The information in the UFSAR supplement, submitted pursuant to 10 CFR 54.21(d), as revised during the license renewal application review process, and as supplemented by the Commitments applicable to Byron Unit 2 in Appendix A of the "Safety Evaluation Report Related to the License Renewal of Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2" (SER) dated July 2015, is collectively the "License Renewal UFSAR Supplement." This

Supplement is henceforth part of the UFSAR which will be updated in accordance with 10 CFR 50.71(e). As such, the licensee may make changes to the programs and activities applicable to Byron Unit 2 described in this Supplement provided the licensee evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

(b) This License Renewal UFSAR Supplement, as revised per License Condition 12(a) above, describes certain programs to be implemented and activities to be completed prior to the period of extended operation.

1. The licensee shall implement those new programs and enhancements to existing programs no later than May 6, 2026.

2. The licensee shall complete those activities as noted in the Commitments applicable to Byron Unit 2 in this Supplement no later than May 6, 2026, or the end of the last refueling outage prior to the period of extended operation, whichever occurs later.

3. The licensee shall notify the NRC in writing within 30 days after having accomplished item (b)1 above and include the status of those activities that have been or remain to be completed in item (b)2 above.

(13) Adoption of 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems, and components for nuclear power plants"

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, Class 3, and non-Code class SSCs and their associated supports; and the results of non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards, i.e., seismic margin analysis (SMA) to evaluate seismic risk, and a screening of other external hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009; as specified in the license amendment No. 204, dated October 22, 2018.

Exelon will complete the updated implementation items listed in Attachment 1 of Exelon letter to NRC dated September 13, 2018, prior to implementation of 10 CFR 50.69. All issues identified in the attachment will be addressed and any associated changes will be made, focused scope peer reviews will be performed on changes that are PRA upgrades as defined in the PRA standard (ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, Revision 2), and any findings will be resolved and reflected in the PRA of record prior to implementation of the 10 CFR 50.69 categorization process.

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

D. The facility requires no exemptions from the requirements of 10 CFR Part 50.

An exemption was previously granted pursuant to 10 CFR 70.24. The exemption was granted with NRC materials license No. SNM-1916, issued March 4, 1985, and relieved the licensee from the requirement of having a criticality alarm system. Therefore, the licensee is exempted from the criticality alarm system provision of 10 CFR 70.24 so far as this section applies to the storage of fuel assemblies held under this license.

E. The licensee shall implement and maintain in effect all provisions of the approved fire protection program as described in the licensee's Fire Protection Report and the licensee's letters dated September 23, 1986, October 23, 1986, November 3, 1986, December 12 and 15, 1986, and January 21, 1987, and as approved in the SER dated February 1982 through Supplement No. 8, subject to the following provision:

> The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

F. Exelon Generation Company shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualifications, 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[1], which contain Safeguards Information protected under 10 CFR 73.21, is entitled: "Byron Nuclear Power Station Security Plan, Training and Qualification Plan, and Safeguards Contingency Plan, Revision 3," submitted by letter dated May 17, 2006.

Exelon Generation Company 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 Exelon Generation Company CSP was approved by License Amendment No.175 and modified by License Amendment No. 191.

G. Deleted

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[1] The training and Qualification Plan and Safeguards Contingency Plan are Appendices to the Security Plan

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATED

TO AMENDMENT NO. 198 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-72,

AMENDMENT NO. 198 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-77,

AMENDMENT NO. 204 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-37,

AND AMENDMENT NO. 204 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-66

EXELON GENERATION COMPANY, LLC

BRAIDWOOD STATION, UNITS 1 AND 2

BYRON STATION, UNIT NOS. 1 AND 2

DOCKET NOS. STN 50-456, STN 50-457,

STN 50-454, AND STN 50-455

1.0 INTRODUCTION

By application dated September 1, 2017 (Reference 1), as supplemented by letters dated April 4, June 13, and September 13, 2018 (References 2, 3, and 27 respectively), Exelon Generation Company, LLC (Exelon, the licensee) submitted a license amendment request (LAR) to change Renewed Facility Operating Licenses NFP-72 and NFP-77 for Braidwood Station, Units 1 and 2; and NFP-37 and NFP-66 for Byron Station, Unit Nos. 1 and 2 (Braidwood/Byron). The licensee proposed to add a new license condition to the Renewed Facility Operating Licenses to allow the implementation of Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.69, "Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors." The provisions of 10 CFR 50.69 allow adjustment of the scope of structures, systems and components (SSCs) subject to special treatment requirements (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation) based on a method of categorizing SSCs according to their safety significance.

By email dated May 9, 2018 (Reference 24), the U.S. Nuclear Regulatory Commission (NRC, the Commission) staff requested additional information (RAI) from the licensee. By letter dated June 13, 2018 (Reference 3), the licensee responded to the request. The supplements dated April 4, June 13, and September 13, 2018, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on September 26, 2017 (82 FR 44854).

2.0 REGULATORY EVALUATION

2.1 Risk-Informed Categorization and Treatment of SSCs

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

To take advantage of the safety enhancements available through the use of PRA, the NRC staff promulgated a new regulation, 10 CFR 50.69, in the *Federal Register* on November 22, 2004 (69 FR 68008), which became effective on December 22, 2004. The provisions of 10 CFR 50.69 allow adjustment of the scope of SSCs subject to special treatment requirements. Special treatment refers to those requirements that provide increased assurance beyond normal industry practices that SSCs perform their design-basis functions. For SSCs categorized as low safety-significance, alternative treatment requirements may be implemented in accordance with the regulation. For SSCs determined to be of high safety-significance, requirements may not be changed. This approach allows improved focus on equipment that has high safety-significance resulting in improved plant safety.

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

Section 50.69 of 10 CFR does not allow for the elimination of SSC functional requirements or allow equipment that is required by the deterministic design basis to be removed from the facility. Instead, 10 CFR 50.69 enables licensees to focus their resources on SSCs that make a significant contribution to plant safety. Electric equipment important to safety relied for accident mitigation, covered within the scope of 10 CFR 50.69, should continue to have demonstrated evidence of environmental qualification that equipment can perform its safety function during and after a design basis accident. For SSCs that are categorized as high safety-significant (HSS), existing treatment requirements are maintained or potentially enhanced. On the other hand, for SSCs categorized as low safety-significant (LSS) that do not significantly contribute to plant safety on an individual basis, the regulation allows an alternative risk-informed approach to treatment that provides a reasonable, although reduced, level of confidence that these SSCs will satisfy functional requirements. Implementation of 10 CFR 50.69 allows licensees to improve focus on equipment that has high safety-significance, resulting in improved plant safety.

2.2 Licensee's Proposed Changes

By letter dated September 13, 2018, 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, Class 3, and non-Code class SSCs and their associated supports; and the results of non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards, i.e., seismic margin analysis (SMA) to evaluate seismic risk, and a screening of other external hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009; as specified in the license amendment No. [XXX], dated [DATE].
>>
>> Exelon will complete the updated implementation items listed in Attachment 1 of Exelon letter to NRC dated September 13, 2018, prior to implementation of 10 CFR 50.69. All issues identified in the attachment will be addressed and any associated changes will be made, focused scope peer reviews will be performed on changes that are PRA upgrades as defined in the PRA standard (ASME/ANS [American Society of Mechanical Engineers/American Nuclear Society] RA-Sa-2009), as endorsed by RG 1.200, Revision 2), and any findings will be resolved and reflected in the PRA of record prior to implementation of the 10 CFR 50.69 categorization process.
>>
>> Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from a seismic margins approach to a seismic probabilistic risk assessment approach).

2.3 Regulatory Review

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

Regulatory Requirements

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

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[4]

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. Additionally, 10 CFR 50.69(g) requires licensee to submit a report under 10 CFR 50.73(b) for any event or condition that prevented, or would have prevented, a RISC-1 or RISC-2 SSC from performing a safety significant function. For LSS SSCs, licensees can implement alternative treatment requirements in accordance with 10 CFR 50.69(b)(1) and 10 CFR 50.69(d). For RISC-3 SSCs, licensees can replace special treatment with an alternative treatment. For RISC-4 SSCs, 10 CFR 50.69 does not impose new treatment requirements, and RISC-4 SSCs are removed from the scope of any applicable special treatment requirements identified in 10 CFR 50.69(b)(1).

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

(i) Consider results and insights from the plant-specific PRA. This PRA must at a minimum model severe accident scenarios resulting from internal initiating events occurring at full power operation. The PRA must be of sufficient quality and level of detail to support the categorization process, 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

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

considered include design bases functions and functions credited for mitigation and prevention of severe accidents. All aspects of the integrated, systematic process used to characterize SSC importance must reasonably reflect the current plant configuration and operating practices, and applicable plant and industry operational experience.

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

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

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

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

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

Guidance

Nuclear Energy Institute (NEI) 00-04, Revision 0, "10 CFR 50.69 SSC Categorization Guideline" (Reference 4), describes a process for determining the safety-significance of SSCs and categorizing them into the four RISC categories defined in 10 CFR 50.69. This categorization process is an integrated decision-making process that incorporates risk and traditional engineering insights. NEI 00-04, Revision 0, provides options for licensees implementing different approaches depending on the scope of their PRA models. It also allows the use of non-PRA approaches when PRAs have not been performed. NEI 00-04 identifies non-PRA approaches such as fire-induced vulnerability evaluation to address fire risk, seismic margin analysis (SMA) to address seismic risk, and guidance in Nuclear Management and Resource Council (NUMARC) 91-06, "Guidelines for Industry Actions to Assess Shutdown Management," (Reference 5), to address shutdown operations. As stated in Regulatory Guide (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 evaluation.

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

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

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

RG 1.201 (Trial Use), Revision 1, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants according to Their Safety Significance" (Reference 6), endorses the categorization method described in NEI 00-04, Revision 0, with clarifications, limitations, and conditions. RG 1.201 states that the applicant is expected to document, at a minimum, the technical adequacy of the internal initiating events PRA. Licensees may use either PRAs or alternative approaches for hazards other than internal initiating events. RG 1.201 clarifies that the NRC staff expects that licensees proposing to use non-PRA approaches in their categorization should provide a basis in the submittal for why the approach and the accompanying method employed to assign safety-significance to SSCs is technically adequate. It further states that as part of the NRC's review and approval of a licensee's or applicant's application requesting to implement 10 CFR 50.69, the NRC staff intends to impose a license condition that will explicitly address the scope of the PRA and non-PRA methods used in the licensee's categorization approach. If a licensee or applicant wishes to change its categorization approach and the change is outside the bounds of the NRC's license condition (e.g., switch from a seismic margins analysis to a seismic PRA), the licensee or applicant will need to seek NRC approval via a license amendment, of the implementation of the new approach in their categorization process. RG 1.201 also states that all aspects of NEI 00-04 must be followed to achieve reasonable confidence in the evaluations required by 10 CFR 50.69(c)(1)(iv).

RG 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities" (Reference 7) describes an acceptable approach for determining whether the quality of the PRA, in total or the parts that are used to support an application, is sufficient to provide confidence in the results such that the PRA can be used in regulatory decision making for light-water reactors. It endorses, with clarifications, the ASME/ANS PRA Standard RA-Sa-2009 ("ASME/ANS 2009 Standard" or "PRA Standard") (Reference 8). This RG provides guidance for determining the technical adequacy of a PRA by comparing the PRA to the relevant parts of the ASME/ANS RA-Sa-2009 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 9), provides guidance on the use of PRA findings and risk insights in support of changes to a plant's licensing basis. This RG provides risk acceptance guidelines for evaluating the results of such evaluations.

3.0 TECHNICAL EVALUATION

3.1 Staff's Method of Review

In determining whether an amendment to a license will be issued, the Commission is guided by the considerations that govern the issuance of initial licenses to the extent applicable and appropriate. The staff evaluated the licensee's application to determine if the proposed changes are consistent with the regulations and guidance discussed in Section 2 of this safety evaluation (SE). Paragraph 50.69(b)(3) of 10 CFR states that the Commission will approve a licensee's implementation of 10 CFR 50.69 by issuing a license amendment if it determines that the licensee's process for categorizing SSCs satisfies the requirements to 10 CFR 50.69(c). The staff reviewed the licensee's SSC categorization process against the categorization process guidance described in NEI 00-04, Revision 0, as endorsed by RG 1.201, and against the requirements in 10 CFR 50.69(c). The staff's review and the documentation of that review in this safety evaluation (SE) uses the framework of NEI 00-04, Revision 0.

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

Sections 1.5 and 2 of NEI 00-04 provide an overview of the categorization process. RG 1.201 provides that the categorization process described in NEI 00-04 with any noted exceptions or clarifications, is acceptable for implementation of 10 CFR 50.69. Categorization is performed system by system. An SSC cannot be categorized if it supports multiple functions unless the process includes provisions ensuring that the SSCs supporting multiple functions over multiple systems are assigned the highest risk significance for any of the associated functions.

The licensee stated in the LAR that it will implement the risk categorization process in accordance with NEI 00-04, as endorsed by RG 1.201; however, the licensee provided little detail of the categorization process. Therefore, in Request for Additional Information (RAI) 05 the NRC staff requested the licensee to: (1) summarize the categorization process, (2) provide the order of the sequence of elements or steps that will be performed, (3) explain the difference between preliminary HSS and assigned HSS, and (4) identify which inputs can and which cannot be changed by the IDP from preliminary HSS to LSS.

In its letter dated June 13, 2018, in response to RAI 05, the licensee summarized the categorization process and described which steps are performed at the component level and which steps are performed at the function level. The licensee explained that the execution sequence of steps/elements of the process does not impact the resulting preliminary categorization because the safety determination of each element of the process is independent of each other.

As summarized in the licensee's response to RAI 05, the categorization process contains the following elements and is summarized in Table 1 below:

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

- Defining system function and assigning components to functions (see Section 3.4 of this SE).

- Risk Characterization. Safety-significance of active components is assessed through a combination of PRA and non-PRA methods, covering all hazards (see Section 3.5 of this SE).

- DID characterization performed in accordance with Section 6 of NEI 00-04 (see Section 3.6 of this SE).

- Passive Characterization. Passive components are not modeled in the PRA and, therefore, a different assessment method is used to assess the safety-significance of these components, as described in Section 3.5.4 of this SE. This process addresses those components that have only a pressure-retaining function and the passive function of active components, such as the pressure/liquid retention of the body of a motor-operated valve.

- Qualitative Characterization. System functions are qualitatively categorized as HSS or LSS based on the seven questions in Section 9.2 of NEI 00-04 (see Section 3.9 of this SE).

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

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

Table 1

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

Element	Categorization Step - NEI 00-04 Section	Evaluation Level	Drives Associated Functions	IDP Change HSS to LSS
Defense-in-Depth	Core Damage – Section 6.1	Function/ Component	Yes	Not Allowed
	Containment – Section 6.2	Component	Yes	Not Allowed
Qualitative Criteria	Considerations – Section 9.2	Function	N/A	Allowable for Considerations
Passive	Passive – Section 4	Segment/ Component	No	Not Allowed

In its letter dated June 13, 2018, the licensee provided further clarification of the allowable considerations for qualitative criteria in the notes of Table 1, as follows:

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

In its letter dated June 13, 2018, in response to RAI 05.b, the licensee explained that consistent with NEI 00-04, the categorization of a component or function is "preliminary" until it has been confirmed by the IDP (see also Section 3.9 of this SE). The licensee stated that a component or function is preliminarily categorized as HSS if any element of the process results in a preliminary HSS determination. This preliminary categorization will be presented to the IDP for review. The IDP will decide the final categorization as further discussed in Section 3.9 of this SE.

In its letter dated June 13, 2018, in Table 1 of the RAI response, and in response to the NRC's staff request in RAI 05.c, the licensee provided clarifications on how some steps of the process are performed at the component level (e.g., all PRA and non-PRA-modeled hazards, containment DID, passive categorization), how some steps are performed at the function level (e.g., qualitative criteria), and how some steps are performed at the function and component level (e.g., shutdown, core damage DID).

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

The NRC staff has evaluated the categorization steps and the associated clarifications provided by the licensee in its letter dated June 13, 2018, Table 1, in response to RAI 05, and finds that the licensee's process is consistent with all aspects of the process in NEI 00-04, as endorsed by RG 1.201.

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

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

Section 4 of NEI 00-04 states that system selection and boundary definition include defining system boundaries where the system interfaces with other systems. NEI 00-04 states that the next step is the identification of system functions, including design basis and beyond design-basis functions identified in the PRA, and that system functions should be consistent with the functions defined in design-basis documentation and maintenance rule functions. NEI 00-04 states that the coarse mapping of components to functions involves the initial breakdown of system components into system functions they support. The licensee should then identify and document system components and equipment associated with each function.

Paragraph 50.69(c)(1)(v) of 10 CFR requires that categorization be performed for entire systems and structures, not for selected components within a system or structure. The process described in the licensee's letter dated September 1, 2017, 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 met upon implementation of the licensee's 10 CFR 50.69 categorization process.

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

In its letter dated September 1, 2017, Section 2.2, the licensee 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). In Section 3.1.1 of its September 1, 2017, letter, the licensee summarizes the different hazards and plant states for which functional and risk-significant information will be collected. Section 3.1.1 also states that the SSC categorization process documentation will include, among other items, system functions identified and categorized with the associated bases and mapping of components to support function(s).

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

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

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

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

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

In its September 1, 2017, letter, Sections 3.1.1 and 3.2.1 through 3.2.5, the licensee explains that its categorization process uses PRA to assess risks from internal events (including internal flooding) and from fire. In the other three risk hazard groups, the licensee's process uses non-PRA methods for the risk characterization, as follows:

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

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

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

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

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

Internal Events and Internal Flooding PRA

The NRC staff reviewed the results of the peer review of the internal events and flooding PRAs and associated facts and observations (F&O) closure review described in LAR Section 3.3 and presented in LAR Attachment 3. The internal events model was subject to a self-assessment and a full scope peer review in July 2013. In its letter dated June 13, 2018, in response to RAI 01, the licensee confirmed that the peer review was performed in accordance with RG 1.200 Revision 2 (Reference 7), for determining the technical adequacy of the PRA and that the review included the internal flooding model. The response to RAI 01 also explained that the self-assessment for both the internal events and internal flooding PRA was performed to assist the licensee preparations for the peer review and, therefore, there are no findings associated with the self-assessment.

The licensee stated that in February 2017, an F&O closure review was performed by an independent team on all internal events finding-level F&Os. This occurred prior to the NRC acceptance of Appendix X (Reference 10), to the guidance in NEI 05-04 (Reference 11), NEI 07-12 (Reference 12), and NEI 12-13 (Reference 13), concerning the process to close out F&Os. The NRC staff accepted, with conditions, a final version of Appendix X in its letter dated May 3, 2017 (Reference 14). By RAI 02, the NRC staff requested confirmation that the F&O closure process was performed in accordance with NRC accepted guidance. In its letter dated June 13, 2018, in response to RAI 02, the licensee provided the necessary details to confirm that the closure review and updated closure report met the accepted guidance. The response provided details that no focused-scope peer review was necessary, that the independent assessment team (IAT) was provided a written assessment of the findings and determined if any resolutions met the ASME/ANS Standard (Reference 8), definition of PRA upgrade, met the criteria for selecting the IAT members, ensured the assessment was performed to CC-II requirements, and that the review encompassed all finding level F&Os.

By RAI 04, the NRC staff requested the licensee state if any model upgrades were performed to resolve any RAIs, to propose a mechanism to ensure a focused peer review occurred before implementing the 10 CFR 50.69 process. By its letter dated June 13, 2018, the licensee stated that no PRA upgrades were implemented in resolving these issues.

By letter dated September 1, 2017, Attachment 3, the licensee submitted all the open F&Os from the peer reviews, i.e., those F&Os that were not considered resolved by the F&O closure review. In each F&O, the licensee provided a disposition.

The NRC staff reviewed the licensee's resolution of all the open peer review findings and assessed the potential impact of the findings on the categorization. Partially resolved F&O SY-B12-01 noted the exclusion of heating ventilation and air conditioning (HVAC) dependency from the high energy line break scenario in the PRA model. In RAI 03.a, the NRC staff observed that this modeling approach could potentially increase the risk importance values for certain system components above the NEI 00-04, Section 5 threshold criteria for determining HSS. Therefore, the NRC staff requested that the licensee provide an explanation for this modeling exclusion. By letter dated June 13, 2018, in response to RAI 03.a, the licensee proposed implementation item 3.a to update the internal events PRA to include the HVAC dependency in these scenarios prior to implementation of the 10 CFR 50.69 categorization (see Section 3.5.5 of this SE).

In its letter dated September 1, 2017, the licensee states that both the Braidwood/Byron sites assessed their internal events PRA models as one as part of a simultaneous peer review. This implied that the same model was not only used for each unit but for each site, therefore, the NRC staff requested clarification and justification in RAI 10 for the use of one model. In its letter dated June 13, 2017, in response to RAI 10, the licensee stated that a majority of the plant components are the same. The licensee also stated that there was one fault tree model for each of the four units and differences between the units were implemented by separate databases and flag files. Flag files are used to determine which parts of the model logic that can be turned 'on' and 'off' by a quantification file to produce site-specific and unit-specific results.

Paragraph 50.69(c)(1)(i) of 10 CFR, requires, in part, that any plant-specific PRA used in the categorization must be of sufficient quality and level of detail to support the process and must be subjected to a peer review process assessed against a standard that is endorsed by the NRC. RG 1.200 provides guidance for determining the technical adequacy of the PRA by comparing the relevant parts of the ASME/ANS RA-Sa-2009 using a peer review process. Based on the NRC review discussed in the above paragraphs, the NRC staff finds that the licensee has followed the guidance in RG 1.200 and submitted the results of the peer review and, therefore, meets the requirement in 10 CFR 50.69(b)(2)(iii). The NRC staff has reviewed the peer review results and the licensee's resolution of the results and finds that the quality and level of detail of the internal events PRA is sufficient to support the categorization of SSCs as required by 10 CFR 50.69(b)(2)(ii) and using the process endorsed by the NRC staff in RG 1.201. Significant errors and weaknesses in the internal events PRA will be resolved prior to implementation of the 10 CFR 50.69 categorization process with the completion of implementation items 3.a (discussed in this section of the SE) and 11 (discussed in Section 3.5.2 of this SE). Therefore, the NRC staff concludes that the internal events PRA with the completion of the proposed implementation items 3.a and 11 meets the internal events PRA requirement in 50.69(c)(1)(i).

<u>FPRA</u>

The NRC staff reviewed the results of the peer review of the FPRA and associated F&O closure review described in the licensee's letter dated September 1, 2017, Section 3.3 and Attachment 3. The licensee's FPRA was subject to a self-assessment and full-scope industry peer review in June 2015. In its letter dated June 13, 2018, in response to RAI 01, the licensee confirmed that the peer review was performed in accordance with RG 1.200, Revision 2

(Reference 7), for determining the technical adequacy of the PRA. The licensee also explained that the self-assessment for the FPRA was performed to assist the licensee to prepare for the peer review and, therefore, there are no findings associated with the self-assessment.

In its letter dated September 1, 2017, the licensee stated that in February 2017, an F&O closure review was performed by an independent team on all fire finding-level F&Os. This occurred prior to the NRC acceptance of the Appendix X (Reference 11), to the guidance in NEI 05-04 (Reference 10), NEI 07-12 (Reference 12), and NEI 12-13 (Reference 13), concerning the process to "Close Out of Facts and Observations." The NRC staff accepted, with conditions, a final version of Appendix X in the letter dated May 3, 2017 (Reference 14). The NRC staff requested confirmation in RAI 02 that the F&O closure process was performed in accordance with NRC accepted guidance. In its letter dated June 13, 2018, in response to RAI 02, the licensee provided the necessary details to confirm that the closure review and updated closure report met the accepted guidance. The response provided details that no focused-scope peer review was necessary, that the IAT was provided a written assessment of the findings and determined if any resolutions met the ASME/ANS Standard (Reference 8) definition of PRA upgrade, met the criteria for selecting the IAT members, ensured the assessment was performed to CC-II requirements, and that review encompassed all finding level F&Os.

The NRC staff requested in RAI 04 that if any model upgrades were performed to resolve any RAI to propose a mechanism to ensure a focused peer review occurred before implementing the 10 CFR 50.69 process. In its letter dated June 13, 2018, in response to RAI 04, the licensee stated that no PRA upgrades were implemented in resolving these issues.

The NRC staff reviewed the licensee's resolution of the open peer review findings and considered the potential impact of the findings on the 10 CFR 50.69 categorization. F&O 16-4 concerning breaker coordination stated that the coordination calculations were not available at the time of the closure review. In RAI 03.b, NRC staff requested a description of the results of the completed breaker coordination study, identification of the circuits that could not be confirmed coordinated, and explanation of how the inadequate circuits would be modeled in the FPRA. In its letter dated June 13, 2018, in response to RAI 03.b, the licensee explained that several breakers associated with the 480 V load centers and all breakers on the 120 VAC instrument buses were found to lack adequate coordination. The licensee clarified that the coordination study was performed in accordance with NUREG/CR-6850 and that cable length was not credited in the study. The licensee proposed implementation item 3.b to incorporate failures associated with the inadequate circuits into the FPRA model prior to implementation of the 10 CFR 50.69 categorization process (see Section 3.5.5 of this SE). The licensee explained that failures associated with the inadequately coordinated circuits will be modeled by failing the entire bus associated with an inadequately coordinated circuit.

The disposition to F&O 20-8 indicated that the approach to crediting alternate shutdown given abandonment of the main control room (MCR) relied on "scaling factors" associated with degrees of fire-induced damage, instead of fault tree modeling involving failure of SSCs that may be the subject of risk-informed categorization. The disposition states that this same approach was used in the FPRAs supporting certain NFPA 805 LARs and was accepted in NRC's SE of these LARs (ADAMS Accession Nos. ML15061A237, ML15344A346, and ML14308A048). In RAI 03.c, the NRC staff requested further explanation of the licensee's treatment of MCR abandonment scenarios and justification that this modeling approach does not impact the 10 CFR 50.69 application. In its letter dated June 13, 2018, in response to RAI 03, the licensee provided a table presenting three multipliers (i.e., 0.1, 0.2, and 1.0) associated with various conditional core damage probabilities (CCDPs) values: less than 0.001

between 0.001 and 0.1, and greater than 0.1. These multipliers were applied to MCR abandonment scenarios as a conservative surrogate for crediting alternate shutdown actions. The licensee proposed implementation item 3.c to conduct a sensitivity study during the 10 CFR 50.69 categorization (see Section 3.5.5 of this SE) to remove the scaling factors. The NRC staff finds that the proposed sensitivity study is consistent with Table 5-3 in NEI 00-04, which states that additional sensitivity studies will be identified. Because this sensitivity study will address the impact of the uncertainties regarding alternate shutdown following MCR abandonment on the 10 CFR 50.69 categorization results, the NRC staff finds the licensee's approach of using scaling factors acceptable for the application.

Open F&O 26-9 concerns improperly screened wall-mounted electrical panels with greater than four switches. In RAI 03.d, the NRC staff requested the licensee to justify that improperly screened electrical panels do not impact the application. In its letter dated June 13, 2017, in response to RAI 03.d, the licensee proposed implementation item 3.d to identify all wall mounted panel configurations with four or more switches and incorporate any required model changes into the FPRA prior to implementation of the 10 CFR 50.69 categorization process (see Section 3.5.5 of this SE).

The disposition associated with F&O 25-11 stated that the modeling of containment sump (e.g., screen) clogging has not been updated in the FPRA model to the most current industry guidance (i.e., Westinghouse Commercial Atomic Power (WCAP)-16362-NP (Reference 25)). The NRC staff requested justification in RAI 03.e that the current treatment does not impact the application. In its letter dated June 13, 2018, in response to RAI 03.e, the licensee proposed implementation item 3.e to update the sump clogging modeling in the FPRA model consistent with WCAP-16362-NP prior to implementation of the 10 CFR 50.69 categorization process (see Section 3.5.5 of this SE).

The NRC staff requested in RAI 03.f, clarification of the disposition associated with F&O 25-5 which concerned joint human error probabilities (HEPs), because it did not appear to match the finding which concerned review of the top scenarios. In its letter dated June 13, 2018, in response to RAI 03.f, the licensee indicated that in reviewing the significant risk contributors it identified modeling requiring refinement (i.e., treatment of joint HEPs) to keep the results from being overly conservative. NRC staff found this explanation to be a satisfactory clarification.

Paragraph 50.69(c)(1)(i) of 10 CFR, requires, in part, that any plant-specific PRA used in the categorization must be of sufficient quality and level of detail to support the categorization process and must be subjected to a peer review process assessed against a standard that is endorsed by the NRC. RG 1.200 provides guidance for determining the technical adequacy of a FPRA by comparing the PRA to the relevant parts of the ASME/ANS RA-Sa-2009 Standard using a peer review process. Based on its review as described above, the NRC staff finds that the licensee has followed the guidance in RG 1.200 and submitted the results of the peer review and, therefore, meets the requirement in 10 CFR 50.69(b)(2)(iii). In its letter dated September 1, 2017, the licensee stated that the FPRA only utilizes methods that have been previously accepted by the NRC. The NRC staff has reviewed the peer review results and the licensee's resolution of the results and finds that the quality and level of detail of the FPRA will be sufficient, once the implementation items are satisfactorily implemented, to support the categorization of SSCs as required by 10 CFR 50.69(b)(2)(ii) and using the process endorsed by the NRC staff in RG 1.201. Significant errors and weaknesses with the FPRA will be resolved with the completion of implementation items 3.b, 3.e, 3.d, and 3.e (discussed in this section of the SE) and 8.c and 11 (discussed in Section 3.5.2 of this SE). Therefore, the NRC staff concludes that the quality of the FPRA with the satisfactory completion of the implementation items 3.b, 3.c, 3.d, 3.e, 8.c, and 11 meets the requirement in 10 CFR 50.69(c)(1)(i).

3.5.2 Importance Measures and Sensitivity Studies

Paragraph 50.69(c)(1)(i) of 10 CFR requires the results and insights from the PRA be used during categorization. These requirements are met, in part, by using importance measures and sensitivity studies, as described in the methodology in NEI 00-04, Section 5.0.

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. Components that have internal events importance measures values exceeding the criteria are assigned HSS. Components that have fire event importance measures exceeding the criteria are assigned preliminary HSS. Integrated importance measures over all PRA modeled hazards are calculated per Section 5.6 of NEI 00-04, and components for which these measures exceed the criteria are assigned preliminary HSS.

The guidance in NEI 00-04 specifies sensitivity studies to be conducted for each PRA model. The sensitivity studies are performed to ensure that assumptions associated with these specific uncertain parameters (i.e., human error, common cause failure, and maintenance probabilities) are not masking the importance of a component. The NEI 00-04 guidance states that any additional "applicable sensitivity studies" from characterization of PRA adequacy should be considered. In its letter dated September 1, 2017, Section 3.7, the licensee describes how it searched for additional issues in the internal events (including internal flooding) PRA that should be evaluated with a sensitivity study. The licensee used the NRC guidance in NUREG-1855, "Guidelines on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision-making," (Reference 15), supplemented with the Electric Power Research Institute (EPRI) Technical Report (TR)-1016737, "Treatment of Parameter and Modeling Uncertainty for Probabilistic Risk Assessments" (Reference 16), to identify sources of uncertainty in the internal events PRA. Key assumptions documented in the licensee's PRA were also evaluated to identify additional sources of model uncertainty that could impact the 10 CFR 50.69 application. The assessment concluded that no additional sensitivity analyses were needed to address internal events PRA model-specific assumptions or sources of uncertainty.

In its letter dated September 1, 2017, Attachment 6, the licensee provided the list of assumptions and sources of modeling uncertainty that were reviewed for the internal events, including internal flooding, and FPRAs and the licensee's disposition. The NRC staff found that for some of the assumptions and modeling uncertainties there was insufficient information provided in the dispositions for NRC staff to conclude that they had minimal impact on the 10 CFR 50.69 application. In its response to RAI 08, the licensee provided the needed information as described below.

In RAI 08.a, concerning the success criteria for diesel generator (DG) cooling fans, the NRC staff requested the basis for assuming the success criteria of one cooling fan, given that two fans are required during high outdoor temperatures. In its letter dated June 13, 2018, in response to RAI 08.a, the licensee explained that the PRA model is consistent with the design basis which requires one main ventilation fan be available for each emergency DG, and that the PRA model fails the DG upon failure of this fan. The licensee confirmed in their response that no other recoveries for the fan failure are modeled. NRC staff finds the license's modeling of the DG cooling fans appropriate because the licensee modeled the as-built, as-operated plant.

In RAI 08.b, concerning operator actions required to support the auxiliary feedwater system after the condensate storage tank (CST) is depleted, the NRC staff requested justification for excluding failure of these operator actions in the PRA models. In its letter dated June 13, 2018, in response to RAI 08.b, the licensee clarified that based on more recent assessments, the CST provides sufficient inventory to the feedwater system for greater than the 24-hour mission time and, therefore, the operator actions are no longer required. The NRC staff finds the licensee's modeling of the feedwater inventory acceptable for the application because the licensee modeled the as-built, as-operated plant, as supported by its assessment of CST inventory.

In RAI 08.c, associated with dependency analysis for post-fire HEP, the NRC staff requested the bases for using minimum joint HEPs ("floor values") lower than 1E-5 in the FPRA. In its letter dated June 13, 2017, in response to RAI 08.c, the licensee stated that it will continue to use a 1E-6 floor value for dependent joint HEP, but proposed implementation item 8.c (see Section 3.5.5 of this SE) to have justification in the FPRA documentation for specific HEP combinations for which a value of less than 1E-5 is used. The NRC staff finds the proposed implementation item acceptable.

In its letter dated September 1, 2017, Attachment 6, the licensee stated that the reactor coolant pump (RCP) seal loss-of-coolant accident (LOCA) is modeled using the Westinghouse Operators Group (WOG) 2000 consensus model (References 17 and 18). The NRC staff noted in a status report related to Order No. EA-12-049 that the Braidwood/Byron Stations (References 19 and 25, respectively), have installed the Westinghouse reactor cooling pump (RCP) SHIELD Passive Thermal Shutdown Seals (SDS) (Generation III). Therefore, in RAI 11, the NRC staff requested the licensee to clarify whether the GEN III RCP SDS were credited in the PRA models and whether the limitations and conditions in the NRC SE approving the modeling method for GEN III SDS (Reference 20) are met. In its letter dated June 13, 2018, in response to RAI 11.a, the licensee clarified that the current internal events and FPRA models include credit for GEN III RCP SDS based guidance in Pressurized Water Reactor Owners Group (PWROG) 14001-P, Revision 1 (Reference 20). The licensee stated that there are three limitations and conditions that impact the PRA model and that two of them are included in the PRA model. Regarding the third one, the licensee proposed implementation item 11 to incorporate the SDS bypass failure mode into the internal events and FPRA models prior to implementation of the 10 CFR 50.69 categorization process (see Section 3.5.5 of this SE).

In RAI 11.b, the NRC staff requested the licensee to clarify whether the implementation of the GEN III RCP SDS model has been peer-reviewed, and if not, justify why not. In response to RAI 11.b.iv, the licensee stated that only the 2000 WOG model has been peer reviewed in 2013, and that the inclusion of the SDS model did not constitute a PRA upgrade that would require a focused-scope peer review. The licensee stated that this determination was based on the following: (1) the SDS model is not a new methodology, (2) there is no change in scope since equipment, dependencies, and accident sequence types remain the same, and (3) no change in modeling capability since the peer reviewed PRA model can still evaluate risk associated with station blackout and total loss of cooling. The licensee asserted that this update is only a change in expected seal leakages using the RCP SDS model. The RCP SDS model has been approved by the NRC (Reference 20), and this is, therefore, an acceptable implementation item.

Based on the above, the NRC staff finds that the licensee searched for, identified, and evaluated sources of uncertainty in its FPRA consistent with the relevant guidance in NUREG-

1855 (Reference 15), and EPRI document TR-1016737 (Reference 16), and, therefore, satisfied the NEI 00-04 guidance to identify additional "applicable sensitivity studies."

3.5.3 Non-PRA Methods

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

As described in its letter dated September 1, 2017, the licensee's categorization process uses the following non-PRA methods:

- SMA to assess seismic risk;
- Screening during the IPEEE to assess risk from other external hazards (high winds, external floods);
- Shutdown safety plan as described in NUMARC 91-06 (Reference 5), to assess shutdown risk.

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

Seismic Risk

To assess seismic risk for the 10 CFR 50.69 categorization process, the licensee proposes to use the SMA method. SMA is a screening method that does not quantify CDF. The licensee performed a seismic margin analysis during the IPEEE (Reference 21). The SMA method includes the development of the seismic safe shutdown equipment list (SSEL) which contains the components that would be needed during and after a seismic event. The SSEL identifies one preferred and one alternate path capable of achieving and maintaining safe shutdown conditions for at least 72 hours following an earthquake. The licensee stated in the LAR that it had updated the IPEEE SSEL to reflect the current as-built and as-operated plant. The licensee further stated that future changes to the plant will be evaluated as needed to determine their impact on the SMA and risk categorization process.

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

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

Other External Hazards (High Winds, External Floods)

As indicated in Section 3.2.4 of the LAR, external hazards were initially evaluated by the licensee during the IPEEE. This hazard category includes all non-seismic external hazards such as high winds, external floods, transportation, nearby facility accidents, and other hazards. The IPEEE external hazard analysis used a progressive screening approach and concluded that all these other hazards are negligible contributors to overall plant risk. Further, the licensee indicated that it had re-evaluated these other external hazards using the criteria in ASME/ANS RA-Sa-2009 PRA Standard (Reference 8). In RAI 06, the NRC staff requested further clarification on the licensee's external hazard analysis. In its letter dated June 13, 2018, the licensee stated in response to RAI 06.a that any hazards that have not been screened in accordance with ASME/ANS RA-Sa-2009 PRA Standard will be evaluated according to the flow chart in Figure 5-6 of NEI 00-04. Additionally, in response to RAI 06.d, the licensee stated that there are no other external hazards that will be evaluated using a method other than depicted in the flow chart in Figure 5-6 of NEI 00-04.

The NRC staff also requested the licensee identify which SSCs contributed to screening extreme winds and tornados from the categorization process. In its letter dated June 13, 2018, in response to RAI 06.e, the licensee identified there are a few SSCs credited in the screening of extreme winds and tornado hazards: the eight service water cooling towers and their supporting SSCs (e.g., fans, riser valves, electrical switchgear). The licensee stated that these SSCs will be considered HSSs.

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

Shutdown Risk

Consistent with the NEI 00-04 guidance endorsed by the NRC, the licensee proposes to use the shutdown safety assessment process based on NUMARC 91-06 (Reference 5). NUMARC 91-06 provides considerations for maintaining DID for the five key safety functions during shutdown. Specifically, these key safety functions are 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.

In its letter dated June 13, 2018, in the licensee's response to RAI 07 and consistent with the guidance in NEI 00-04, Section 5.5, the licensee indicated that components are categorized with respect to shutdown risk using a non-PRA shutdown assessment as follows:

- If a system/train supports a key safety function as the primary or first alternate means, then it is considered to be a "primary shutdown safety system" and is categorized as preliminary HSS. NEI 00-04 defines a "primary shutdown safety system" as also having the following attributes:
  - a technical basis for its ability to perform the function.
  - a margin to fulfill the safety function.
  - does not require extensive manual manipulation to fulfill its safety function.
- If the SSC's failure would initiate an event during shutdown plant conditions (e.g., loss of shutdown cooling, drain down), then that SSC is categorized as preliminary HSS.

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

3.5.4 Component Safety-Significance Assessment for Passive Components

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

In its letter dated September 1, 2017, Section 3.1.2, the licensee proposed using a categorization method for passive components not cited in NEI 00-04 for passive component categorization, but approved by the NRC for use at ANO-2 (Reference 22). 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 23). ASME Code Case N-660 has been incorporated by reference into 10 CFR 50.55a, with conditions, by inclusion in RG 1.147, Revision 18, dated March 2017 (ADAMS Accession No. ML16321A336). The ANO-2 methodology relies on the conditional core damage and large early release probabilities associated with pipe ruptures. Safety-significance is generally measured by the frequency and the consequence of, in this case, pipe ruptures. Treatment requirements (including repair/replacement) only affect the frequency of passive component failure. Categorizing solely based on consequences, which measures the safety-significance of the pipe given that it ruptures, is conservative compared to including the rupture frequency in the categorization. The categorization will not be affected by changes in frequency arising from changes to the treatment. Therefore, the NRC staff finds that the use of the repair/replacement methodology is acceptable and appropriate for passive component categorization of Class 2 and Class 3 SSCs.

In its letter dated September 1, 2017, the licensee did not specify what class of passive components will be categorized with the ANO-2 methodology. The NRC staff noted that since Class 1 SSCs constitute principal fission product barriers as part of the reactor coolant system or containment, the consequence of pressure boundary failure for Class 1 SSCs may be different than for Class 2 and Class 3 and, therefore, the criteria in the ANO-2 methodology cannot automatically be generalized to Class 1 SSCs without further justification. A technical justification for Class 1 SSCs would have to address how the methodology is sufficiently robust to assess the safety significance of Class 1 SSCs. This justification would have to include, but not be limited to, the following: (1) justification of the appropriateness of the conditional core damage probability numerical criteria used to assign 'high', 'medium' and 'low' safety significance to these loss of coolant initiating events; (2) identification and justification of the adequacy of the additional qualitative considerations to assign 'medium' safety-significance (based on the conditional core damage probability) to 'high' safety significance; (3) justification for crediting operator actions for success and failure of pressure boundary; and (4) guidelines and justification for selecting the appropriate break size (e.g., double-ended guillotine break or smaller break). The justification would also need to include supporting examples of types of Class 1 SSCs that would be assigned low safety significance. Therefore, in RAI 09 the NRC staff requested the licensee to either confirm that only Class 2 and Class 3 SSCs will be categorized using ANO-2 passive methodology or to explain and justify how the methodology will be modified to include Class 1 components.

In its letter dated June 13, 2018, in response to RAI 09, the licensee stated that it will apply the process for the passive categorization of Class 2, Class 3, and non-Code class components, and that all ASME Code Class 1 SSCs with a pressure retaining function, as well as supports, will be designated as HSS for the passive categorization. The licensee further clarified that this HSS designation for Class 1 SSCs cannot be changed by the IDP. In the response to RAI 09, the licensee included non-Code class component piping for which the pressure retaining function can be categorized. The use of risk insights to determine treatment options for passive functions of non-Code class components is consistent with the use of risk insights to determine treatment options for non-safety related SSCs and therefore the inclusion of non-code class components is acceptable. Because all Class 1 SSCs and supports will be considered HSS and only Class 2, Class 3, non-Code class SSCs will be categorized using the ANO-2 passive categorization methodology consistent with previous NRC staff approval, the NRC staff finds the licensee's proposed approach for passive categorization acceptable for the 10 CFR 50.69 categorization process.

3.5.5 Summary

The NRC staff reviewed the PRA and the non-PRA approaches and 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 (Reference 3) and the NRC-endorsed guidance in NEI 00-04 (Reference 4). Accordingly, subject to the proposed license condition described below, the staff approves the use of the following methods in the licensee's 10 CFR 50.69 categorization process:

- PRA to assess internal events, including internal flooding risk
- FPRA to assess fire risk
- SMA to assess seismic risk
- Screening using IPEEE to assess risk from other external hazards (high winds, external floods)
- Shutdown safety plan to assess shutdown risk
- ANO-2 passive categorization method (see Reference 21) to assess passive component risk for Class 2, Class 3, and non-Code SSCs and their associated supports

Based on its review of the LAR and the licensee's responses to the NRC staff's RAIs, the staff identified certain specific actions necessary to support its conclusion that the proposed program meets the requirements in 10 CFR 50.69 and the guidance in RG 1.201 and NEI 00-04. In its letter dated September 13, 2018, the licensee proposed the addition of a license condition for the implementation of 10 CFR 50.69 (see Section 4.0 of this SE). Specifically, the license condition (Reference 27), identifies six implementation items that will be completed prior to the implementation of the 10 CFR 50.69 categorization process, and one additional sensitivity study that will be completed as part of the categorization process:

3.a The internal events and FPRA models will be updated to model HVAC dependency for HELB scenarios prior to implementation of the 10 CFR 50.69 risk categorization process.

3.b The FPRA models for Braidwood/Byron will be updated to incorporate failures required to account for instances where breaker coordination cannot be confirmed prior to implementation of the 10 CFR 50.69 risk categorization process.

3.c To ensure that the impact of the CCDP and conditional large early release probability (CLERP) scaling factor adjustments, used for crediting alternate shutdown given abandonment of the MCR, is accounted for in the categorization process, an FPRA sensitivity that removes the scaling factor adjustments will be performed during the 10 CFR 50.69 categorization process, in addition to the sensitivities required by NEI 00-04 Table 5-3. If the FPRA is updated in the future to eliminate the scaling factor adjustment, this sensitivity calculation would no longer be required.

3.d Identification of all wall mounted panel configurations with four or more switches will be completed and any resulting changes to Braidwood/Byron FPRA models to incorporate the impact of these panels will be made prior to implementation of the 10 CFR 50.69 risk categorization process.

3.e The Braidwood/Byron FPRA models that will be used for 10 CFR 50.69 implementation will include a new sump clogging value consistent with the WCAP-16362-NP guidance.

8.c The Braidwood/Byron FPRAs to be used to support the implementation of the 50.69 categorization will retain a 1 E-06 joint HEP floor value and justification will be included in the FPRA documentation for specific HEP combinations for which a value of less than 1 E-05 is used.

11 The additional failure contribution of the Westinghouse RCP shutdown seal bypass failure mode will be added to the Braidwood/Byron internal events and FPRA models, consistent with the limitations and conditions in the NRC SE for PWROG-140001-P, Revision 1 (ADAMS Accession No. ML17200A116), prior to implementation of the 10 CFR 50.69 risk categorization process.

3.6 DID (NEI 00-04, Section 6)

NEI 00-04, Section 6.0, provides guidance on assessment of DID. Figure 6-1 in NEI 00-04 provides guidance to assess design-basis DID based on the likelihood of the design-basis internal event initiating event and the number of redundant and diverse trains nominally available to mitigate the initiating event. The likelihood of the initiating events is binned and, for different likelihood bins, HSS is assigned if fewer than the indicated number of mitigating trains are nominally available. Section 6 also provides guidance to assess containment DID based on preserving containment isolation and long-term containment integrity and on preventing containment bypass and early hydrogen burns. DID for beyond design-basis initiating events is addressed by the PRA categorization process.

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

Section 6 indicates that the safety-significance determined by the guidance is HSS, and the licensee clarified in its letter dated September 1, 2017, Section 3.1.1, that it will require an SSC categorized as HSS based on the DID assessment in Section 6 to be categorized as HSS. Based on its review as described above, the NRC staff finds the licensee's categorization process is consistent with the NRC-endorsed NEI 00-04 guidance and fulfills the 10 CFR 50.69(c)(1)(iii) criteria so that DID is maintained.

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

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

In its letter dated September 1, 2017, Section 3.1.1, the licensee stated that if any component is identified as HSS from either the integrated PRA component safety-significance assessment (Section 5 of NEI 00-04) or the DID assessment (Section 6 of NEI 00-04), the associated system function(s) would be identified as HSS. Once a system function is identified as HSS, all the components that support that function are categorized as preliminary HSS. In RAI 05.d, the NRC staff requested the licensee to clarify whether all aspects identified in Sections 5 and 6 of NEI 00-04, including if any components identified as HSS through Sections 5.3 to 5.5 of NEI 00-04 (dedicated to seismic, external hazards, or shutdown risk) will drive the system functions to be categorized as HSS. In its letter dated June 13, 2018, in response to RAI 05.d, the licensee explained that the safety-significance of functions will be categorized as preliminary HSS only if it is supported by a component determined to be HSS from a PRA-based assessment (i.e., for Braidwood/Byron, internal events and integrated PRA importance measures described in Section 5.6 of NEI 00-04). Components that are identified as HSS from using the non-PRA approaches (SMA, shutdown risk, other external hazards) will not drive the system function(s) they support to be assigned HSS. The licensee explained that non-PRA-based assessments result in the default categorization of any components associated with the safe shutdown success paths defined in those deterministic assessments to be HSS, regardless of its risk significance. The licensee referenced Section 7.1 of NEI 00-04, endorsed without comment in RG 1.201, which states:

> If any SSC is safety significant, from either the PRA-based component safety-significance assessment (Section 5) or the defense-in-depth assessment (Section 6), then the associated system function is preliminarily safety significant. All other functions/SSCs can be preliminarily assigned low safety-significance.

The NRC staff finds that the above description is consistent with NEI 00-04, and therefore, acceptable.

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

Paragraph 50.69(c)(1)(iv) of 10 CFR requires, in part, that any potential increases in CDF and LERF resulting from changes to treatment are small. The categorization process described in the NRC-endorsed NEI 00-04 guidance includes an overall risk sensitivity study for all the LSS components to confirm that if the unreliability of the components were increased, the increase in risk would be small (i.e., meet the acceptance guidelines of RG 1.174). In its letter dated September 1, 2017, Sections 3.1.1 and 3.2.7, the licensee clarifies that in the sensitivity study, the unreliability of all LSS SSCs modeled in the PRA(s) will be increased by a factor of 3.

Separate sensitivity studies are to be performed for each system categorized as well as a cumulative sensitivity study for all the SSCs categorized through the 10 CFR 50.69 process.

This sensitivity study, together with the periodic review process discussed in Section 3.11 of this SE, assure that the potential cumulative risk increase from the categorization is small. The performance monitoring process monitors the component performance to ensure that potential increases in failure rates of categorized components are detected and addressed before reaching the rate assumed in the sensitivity study. Based on the above, the NRC staff finds that the licensee will perform the risk sensitivity studies consistent with the guidance in NEI 00-04, Section 8.0 and, therefore, will assure that the potential cumulative risk increase from the categorization is small, as required by 10 CFR 50.69(c)(1)(iv).

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

Section 50.69(c)(2) of 10 CFR requires that the SSCs must be categorized by an IDP staffed with expert, plant-knowledgeable members whose expertise includes, at a minimum, PRA, safety analysis, plant operations, design engineering, and system engineering. In its letter dated September 1, 2017, Section 3.1.1, the licensee clarifies that the IDP will be composed of a group of at least five experts who collectively have expertise in plant operation, design (mechanical and electrical) engineering, system engineering, safety analysis, and PRA.

The guidance in NEI 00-04, endorsed in RG 1.201, 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). As provided by the NEI 00-04 guidance, and as indicated in the licensee's letter dated September 1, 2017, Attachment 1, the process used by the IDP for the categorization of SSCs will be described and documented in a plant procedure.

In its letter dated September 1, 2017, Section 3.1.1, the licensee states that at least three members of the IDP will have a minimum of 5 years of experience at the plant, and there will be at least one member of the IDP who has a minimum of 3 years of experience in modeling and updating of the plant-specific PRA. It further clarifies that the IDP will be trained in the specific technical aspects and requirements related to the categorization process. Training will address, at a minimum, the purpose of the categorization; present treatment requirements for SSCs, including requirements for design-basis events; PRA fundamentals; details of the plant-specific PRA, including the modeling, scope, and assumptions; the interpretation of risk importance measures, and the role of sensitivity studies and the change-in-risk evaluations; and the DID philosophy and requirements to maintain this philosophy.

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

In its letter dated June 13, 2018, the licensee explained in response to RAI 05 that the IDP's authority to change component categorization from preliminary HSS to LSS is limited. The licensee summarized these limitations in Table 1 of the response to RAI 05 as further revised in response to follow-up RAI 05.01. As shown above in Section 3.2 of this SE (Table 1), and consistent with the guidance in NEI 00-04, components found to be HSS from the following aspects of the process cannot be re-categorized by the IDP:

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

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

The IDP will additionally assess the safety significance of functions using the seven qualitative questions/considerations provided in Section 9.2 of NEI 00-04. In its letter dated June 13, 2018, in response to RAI 05, the licensee stated that if the IDP determines that any one of the seven considerations cannot be confirmed (false response) for a function, then the final categorization of that function is HSS.

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

Paragraph 50.69(c)(1)(iv) of 10 CFR requires, in part, reasonable confidence that sufficient safety margins are maintained for SSCs categorized as RISC-3. Safety margins are addressed through an integrated engineering evaluation that would be assessed by the IDP. As discussed in NEI 00-04, the LSS SSC requirements that are relaxed for RISC-3 (LSS) SSCs are those related to treatment, not design or capability, and paragraph 10 CFR 50.69(d)(2)(i) requires that the licensee ensure, with reasonable confidence, that RISC-3 SSCs remain capable of performing their safety-related functions under design basis conditions.

Based on the above, the NRC staff finds that the proposed Braidwood/Byron categorization process with the license condition and implementation items provided in Section 4.0 of this SE are consistent with the endorsed guidance in NEI 00-04 and, therefore, fulfills the 10 CFR 50.69 (c)(1)(iv) criteria that sufficient safety margins are maintained.

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

Paragraph 50.69(c)(1)(ii) of 10 CFR requires, in part, that all aspects of the integrated, systematic process used to characterize SSC importance must reasonably reflect the current plant configuration operating practices, and applicable plant and industry operating experience.

NEI 00-04, Section 11, provides guidance on program documentation and change control and Section 12 provides guidance on periodic review. These sections are described in NEI 00-04 with respect to satisfying 10 CFR 50.69(e) and 10 CFR 50.69(f), respectively. Maintaining change control and periodic review will also maintain confidence that all aspects of the program reflect current plant operation.

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

In its letter dated September 1, 2017, Section 3.2.6, the licensee described administrative controls in place to ensure that the PRA models used to support the categorization reflect the as-built, as-operated plant over time. The licensee's process includes regularly scheduled and interim (as needed) PRA model updates. The process includes provisions for monitoring issues affecting the PRA models (e.g., due to changes in the plant, errors or limitations identified in the model, industry operational experience), for assessing the risk impact of unincorporated changes, and for controlling the model and associated computer files. The process also includes reevaluating previously categorized systems to ensure the continued validity of the categorization. Routine PRA updates are performed every two refueling cycles at a minimum. The NRC staff finds that this description is consistent with the requirements for feedback and process adjustment required by 10 CFR 50.69(e), and is, therefore, acceptable.

Section 50.69(f) of 10 CFR requires program documentation, change control, and records. In its letter dated September 1, 2017, 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 states that the RISC categorization process documentation will include the following ten elements:

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

In addition, the licensee's letter dated September 1, 2017, Attachment 1 (List of Categorization Prerequisites) states that it will establish procedures prior to the use of the categorization process that will contain the following elements: (1) IDP member qualification requirements, (2) qualitative assessment of system functions, (3) component safety-significance assessment, (4) assessment of DID and safety margin, (5) review by the IDP and final determination of safety-significance for system functions and components, (6) risk sensitivity studies to confirm that the risk acceptance guidelines of RG 1.174 are met, (7) periodic review to ensure continued categorization validity and acceptable performance for SSCs that have been categorized, and (8) documentation requirements identified in Section 3.1.1. Procedures are formal plant documents and changes will be tracked providing change control and records of the changes.

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

Based on its evaluation, the NRC staff finds that the change control and performance monitoring of categorized SSCs and PRA updates will sufficiently capture and evaluate component failures to identify significant changes in the failure probabilities. In addition, the PRA update program and associated re-evaluation of component importance will appropriately consider the effects of changing failure probabilities and changing plant configuration on the component safety-significant categories. As discussed above, the 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. Additionally, as a part of 10 CFR 50.69(g), the licensee shall submit a licensee event report under 10 CFR 50.73(b) for any event or condition that prevented or would have prevented a RISC-1 or RISC-2 SSC from performing a safety significant function. Therefore, the process used to characterize SSC importance will reasonably reflect the current plant configuration and operating practices, and applicable plant and industry operational experience required in 10 CFR 50.69(c)(1)(ii).

3.11 Technical Conclusion

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

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

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

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

(4) includes evaluations that provide reasonable confidence that for SSCs categorized as RISC-3, sufficient safety margins are maintained and that any potential increases in CDF

and LERF resulting from changes in treatment are small, as reviewed in Sections 3.8 and 3.9 of this SE and, therefore, meets the requirements in 10 CFR 50.69(c)(1)(iv);

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

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

4.0 10 CFR 50.69 IMPLEMENTATION LICENSE CONDITION

Section 50.69(b)(2) of 10 CFR requires the licensee to submit an application that describes the categorization process. Section 50.69(b)(3) of 10 CFR states that the Commission will approve the license application if it determines that the categorization process satisfies the requirements of 10 CFR 50.69(c). As described in this SE, the NRC staff has concluded that the 10 CFR 50.69 categorization process described in the licensee's application satisfies the requirements of 10 CFR 50.69(c). However, based on its review of the LAR and the licensee's responses to the NRC staff's RAIs, the NRC staff identified certain specific actions, as described below, that are necessary to support the staff's conclusion that the proposed program meets the requirements in 10 CFR 50.69 and the guidance in RG 1.201 and NEI 00-04.

The NRC staff's finding on the acceptability of the PRA evaluation in the licensee's proposed 10 CFR 50.69 process is conditioned on the completion of six changes to the PRA and the addition of one sensitivity study to the studies summarized in Table 5-3 of NEI 00-04. These seven changes are identified in response to RAI 12 (Reference 3). The staff notes that the licensee described some additional minor changes to the PRA and PRA methods. The staff determined that these minor changes would not impact the 10 CFR 50.69 categorization process, and were similar to future changes to the PRA and PRA methods that occur over time. Therefore, the staff determined that these changes do not need to be resolved prior to implementation of the 10 CFR 50.69 process and, therefore, can be addressed and resolved using the licensee's periodic review process.

In its letter dated September 13, 2018 (Reference 27), the licensee proposed the following condition to its license:

> 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, Class 3, and non-Code class SSCs and their associated supports; and the results of non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards, i.e., seismic margin analysis (SMA) to evaluate seismic risk, and a screening of other external hazards updated using the external hazard screening significance process identified in ASME/ANS PRA

Standard RA-Sa-2009; as specified in the license amendment No. [XXX], dated [DATE].

Exelon will complete the updated implementation items listed in Attachment 1 of Exelon letter to NRC dated September 13, 2018, prior to implementation of 10 CFR 50.69. All issues identified in the attachment will be addressed and any associated changes will be made, focused scope peer reviews will be performed on changes that are PRA upgrades as defined in the PRA standard (ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, Revision 2), and any findings will be resolved and reflected in the PRA of record prior to implementation of the 10 CFR 50.69 categorization process.

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

In its September 13, 2018, letter (Reference 27), the licensee proposed the following implementation items:

3.a The internal events and FPRA models will be updated to model HVAC dependency for HELB scenarios prior to implementation of the 10 CFR 50.69 risk categorization process.

3.b The FPRA models for Braidwood/Byron will be updated to incorporate failures required to account for instances where breaker coordination cannot be confirmed prior to implementation of the 10 CFR 50.69 risk categorization process.

3.c To ensure that the impact of the CCDP and CLERP scaling factor adjustments, used for crediting alternate shutdown given abandonment of the MCR, is accounted for in the categorization process, an FPRA sensitivity that removes the scaling factor adjustments will be performed during the 10 CFR 50.69 categorization process, in addition to the sensitivities required by NEI 00-04 Table 5-3. If the FPRA is updated in the future to eliminate the scaling factor adjustment, this sensitivity calculation would no longer be required.

3.d Identification of all wall mounted panel configurations with four or more switches will be completed and any resulting changes to the Braidwood/Byron FPRA models to incorporate the impact of these panels will be made prior to implementation of the 10 CFR 50.69 risk categorization process.

3.e The Braidwood/Byron FPRA models that will be used for 10 CFR 50.69 implementation will include a new sump clogging value consistent with the WCAP- 6362-NP guidance.

8.c The Braidwood/Byron FPRAs to be used to support the implementation of the 50.69 categorization will retain a 1 E-06 joint HEP floor value and justification will be included in the FPRA documentation for specific HEP combinations for which a value of less than 1 E-05 is used.

11 The additional failure contribution of the Westinghouse RCP shutdown seal bypass failure mode will be added to the Braidwood/Byron internal events and

FPRA models, consistent with the limitations and conditions in the NRC SE for PWROG-140001-P, Revision 1 (ADAMS Accession No. ML17200A116), prior to implementation of the 10 CFR 50.69 risk categorization process.

Based on its evaluation in this SE, the NRC staff finds that the proposed license condition and its implementation items are acceptable because they adequately implement 10 CFR 50.69 using models, methods, and approaches consistent with the applicable guidance that has previously been endorsed as acceptable by the NRC. In each implementation item, the licensee and the NRC staff have reached a satisfactory resolution involving the level of detail and main attributes that each remaining item will incorporate into the program upon its completion. The NRC, during future inspections, may choose to examine the closure of the implementation items with the expectation that any variations discovered during this review, or concerns regarding adequate completion of the implementation item, would be tracked and dispositioned appropriately under the licensee's corrective action program, and could be subject to appropriate NRC enforcement action, as completion of the implementation items would be required by the proposed license conditions.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the NRC staff notified the Illinois State official on September 20, 2018, of the proposed issuance of the amendments. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20 or change inspections or surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding, which was published in the *Federal Register* on November 21, 2017 (82 FR 55404), that the amendments involve no significant hazards consideration, and there has been no public comment on such finding. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

7.0 CONCLUSION

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

8.0 REFERENCES

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Date of issuance: October 22, 2018

SUBJECT: BRAIDWOOD 1 & 2, BYRON 1 & 2 - ISSUANCE OF AMENDMENTS NOS 198, 198, 204, AND 204, RESPECTIVELY, REGARDING ADOPTION OF TITLE 10 OF THE CODE OF FEDERAL REGULATIONS SECTION 50.69, "RISK-INFORMED CATEGORIZATION AND TREATMENT OF STRUCTURES, SYSTEMS AND COMPONENTS FOR NUCLEAR POWER REACTORS" (CAC NOS. MG0201, MG0202, MG0203, AND MG204; EPID L-2017-LLA-0285) DATED OCTOBER 22, 2018

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Accession No. ML18264A092 ***via memo **via e-mail**

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