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

August 28, 2019

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 STATION, UNITS 1 AND 2, AND BYRON STATION, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENTS NOS. 202 and 208 RE: LIMITING CONDITION OF OPERATION FOR INOPERABILITY OF SNUBBERS (EPID L-2019-LLA-0023)

Dear Mr. Hanson:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 202 to Renewed Facility Operating License No. NPF-72 and Amendment No. 202 to Renewed Facility Operating License No. NPF-77 for the Braidwood Station, Units 1 and 2, respectively, and Amendment No. 208 to Renewed Facility Operating License No. NPF-37 and Amendment No. 208 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 January 31, 2019 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19032A149), as supplemented by letter dated August 9, 2019 (ADAMS Accession No. ML19221B562).

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. 202 to NPF-72
2. Amendment No. 202 to NPF-77
3. Amendment No. 208 to NPF-37
4. Amendment No. 208 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. 202
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 January 31, 2019, as supplemented by letter dated August 9, 2019, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-72 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 202 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 30 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Lisa M. Regner, Acting Branch Chief
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications and Renewed Facility Operating License

Date of Issuance: August 28, 2019

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. 202
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 January 31, 2019, as supplemented by letter dated August 9, 2019, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-77 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 202 and the Environmental Protection Plan contained in Appendix B, both of which are attached to Renewed License No. NPF-72, dated January 27, 2016, are hereby incorporated into the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 30 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Lisa M. Regner, Acting Branch Chief
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications and Renewed Facility Operating License

Date of Issuance: August 28, 2019

ATTACHMENT TO LICENSE AMENDMENT NOS. 202 AND 202

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 and Appendix A Technical Specifications 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 Page 3	License NPF-72 Page 3
License NPF-77 Page 3	License NPF-77 Page 3
TSs 3.0 – 1 3.0 – 4	TSs 3.0 – 1 3.0 – 4

(2) Exelon Generation Company, pursuant to the Act and 10 CFR Part 70, to receive, possess and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

(3) Exelon Generation Company, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

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

(5) Exelon Generation Company, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

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

(1) Maximum Power Level

The licensee is authorized to operate the facility at reactor core power levels not in excess of 3645 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 202 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(2) Exelon Generation Company, LLC, pursuant to the Act and 10 CFR Part 70, to receive, possess and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

(3) Exelon Generation Company, LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

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

(5) Exelon Generation Company, LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

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

(1) Maximum Power Level

The licensee is authorized to operate the facility at reactor core power levels not in excess of 3645 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 202 and the Environmental Protection Plan contained in Appendix B, both of which are attached to Renewed License No. NPF-72, dated January 27, 2016, are hereby incorporated into the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

LCO 3.0.1	LCOs shall be met during the MODES or other specified conditions in the Applicability, except as provided in LCO 3.0.2, LCO 3.0.7, and LCO 3.0.9.
LCO 3.0.2	Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5 and LCO 3.0.6. If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required unless otherwise stated.
LCO 3.0.3	When an LCO is not met and the associated ACTIONS are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS, the unit shall be placed in a MODE or other specified condition in which the LCO is not applicable. Action shall be initiated within 1 hour to place the unit, as applicable, in: a. MODE 3 within 7 hours; b. MODE 4 within 13 hours; and c. MODE 5 within 37 hours. Exceptions to this Specification are stated in the individual Specifications. Where corrective measures are completed that permit operation in accordance with the LCO or ACTIONS, completion of the actions required by LCO 3.0.3 is not required. LCO 3.0.3 is only applicable in MODES 1, 2, 3, and 4.

3.0 LCO Applicability

LCO 3.0.7	Exception LCOs allow specified Technical Specification (TS) requirements to be changed to permit performance of special tests and operations. Unless otherwise specified, all other TS requirements remain unchanged. Compliance with Exception LCOs is optional. When an Exception LCO is desired to be met but is not met, the ACTIONS of the Exception LCO shall be met. When an Exception LCO is not desired to be met, entry into a MODE or other specified condition in the Applicability shall be made in accordance with the other applicable Specifications.
LCO 3.0.8	LCOs, including associated ACTIONs, shall apply to each unit individually, unless otherwise indicated. Whenever the LCO refers to a system or component that is shared by both units, the ACTIONs will apply to both units simultaneously.
LCO 3.0.9	When one or more required snubbers are unable to perform their associated support function(s), any affected supported LCO(s) are not required to be declared not met solely for this reason if risk is assessed and managed, and: a. the snubbers not able to perform their associated support function(s) are associated with only one train or subsystem of a multiple train or subsystem supported system or are associated with a single train or subsystem supported system and are able to perform their associated support function within 72 hours; or b. the snubbers not able to perform their associated support function(s) are associated with more than one train or subsystem of a multiple train or subsystem supported system and are able to perform their associated support function within 12 hours. At the end of the specified period the required snubbers must be able to perform their associated support function(s), or the affected supported system LCO(s) shall be declared not met.

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. 208
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 January 31, 2019, as supplemented by letter dated August 9, 2019, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-37 is hereby amended to read as follows:

(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A as revised through Amendment No. 208 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this renewed license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 30 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Lisa M. Regner, Acting Branch Chief
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications and Renewed Facility Operating License

Date of Issuance: August 28, 2019

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. 208
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 January 31, 2019, as supplemented by letter dated August 9, 2019, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-66 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A (NUREG-1113), as revised through Amendment No. 208, and the Environmental Protection Plan contained in Appendix B, both of which were attached to Renewed License No. NPF-37, dated November 19, 2015, are hereby incorporated into this renewed license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 30 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Lisa M. Regner, Acting Branch Chief
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications and Renewed Facility Operating License

Date of Issuance: August 28, 2019

ATTACHMENT TO LICENSE AMENDMENT NOS. 208 AND 208

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 and Appendix A Technical Specifications 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 Page 3	License NPF-37 Page 3
License NPF-66 Page 3	License NPF-66 Page 3
TSs 3.0 – 1 3.0 – 4	TSs 3.0 – 1 3.0 – 4

(2) Pursuant to the Act and 10 CFR Part 70, to receive, possess and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Updated Final Safety Analysis Report, as supplemented and amended;

(3) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

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

(5) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

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

(1) Maximum Power Level

The licensee is authorized to operate the facility at reactor core power levels not in excess of 3645 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 208 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this renewed license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Deleted.

(4) Deleted.

(2) Pursuant to the Act and 10 CFR Part 70, to receive, possess and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Updated Final Safety Analysis Report, as supplemented and amended;

(3) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

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

(5) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

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

(1) <u>Maximum Power Level</u>

The licensee is authorized to operate the facility at reactor core power levels not in excess of 3645 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.

(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A (NUREG-1113), as revised through Amendment No. 208, and the Environmental Protection Plan contained in Appendix B, both of which were attached to Renewed License No. NPF-37, dated November 19, 2015, are hereby incorporated into this renewed license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

LCO 3.0.1 LCOs shall be met during the MODES or other specified conditions in the Applicability, except as provided in LCO 3.0.2, LCO 3.0.7, and LCO 3.0.9.

LCO 3.0.2 Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5 and LCO 3.0.6.

If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required unless otherwise stated.

LCO 3.0.3 When an LCO is not met and the associated ACTIONS are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS, the unit shall be placed in a MODE or other specified condition in which the LCO is not applicable. Action shall be initiated within 1 hour to place the unit, as applicable, in:

a. MODE 3 within 7 hours;

b. MODE 4 within 13 hours; and

c. MODE 5 within 37 hours.

Exceptions to this Specification are stated in the individual Specifications.

Where corrective measures are completed that permit operation in accordance with the LCO or ACTIONS, completion of the actions required by LCO 3.0.3 is not required.

LCO 3.0.3 is only applicable in MODES 1, 2, 3, and 4.

3.0 LCO Applicability

LCO 3.0.7	Exception LCOs allow specified Technical Specification (TS) requirements to be changed to permit performance of special tests and operations. Unless otherwise specified, all other TS requirements remain unchanged. Compliance with Exception LCOs is optional. When an Exception LCO is desired to be met but is not met, the ACTIONS of the Exception LCO shall be met. When an Exception LCO is not desired to be met, entry into a MODE or other specified condition in the Applicability shall be made in accordance with the other applicable Specifications.
LCO 3.0.8	LCOs, including associated ACTIONs, shall apply to each unit individually, unless otherwise indicated. Whenever the LCO refers to a system or component that is shared by both units, the ACTIONs will apply to both units simultaneously.
LCO 3.0.9	When one or more required snubbers are unable to perform their associated support function(s), any affected supported LCO(s) are not required to be declared not met solely for this reason if risk is assessed and managed, and: a. the snubbers not able to perform their associated support function(s) are associated with only one train or subsystem of a multiple train or subsystem supported system or are associated with a single train or subsystem supported system and are able to perform their associated support function within 72 hours; or b. the snubbers not able to perform their associated support function(s) are associated with more than one train or subsystem of a multiple train or subsystem supported system and are able to perform their associated support function within 12 hours. At the end of the specified period the required snubbers must be able to perform their associated support function(s), or the affected supported system LCO(s) shall be declared not met.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATED

TO AMENDMENT NO. 202 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-72,

AMENDMENT NO. 202 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-77,

AMENDMENT NO. 208 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-37,

AND AMENDMENT NO. 208 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 January 31, 2019 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19032A149), as supplemented by letter dated August 9, 2019 (ADAMS Accession No. ML19221B562), Exelon Generation Company, LLC (Exelon, the licensee) submitted a license amendment request for Braidwood Station, Units 1 and 2 (Braidwood) and Byron Station, Unit Nos. 1 and 2 (Byron). The supplement dated August 9, 2019, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the Federal Register on May 7, 2019 (84 FR 19970).

The amendments would revise the technical specifications (TSs) for Braidwood and Byron by adding a new limiting condition for operation (LCO) 3.0.9 to address conditions where one or more snubbers are unable to perform their associated support function. A conforming change would also be made to TS LCO 3.0.1 for Braidwood and Byron to reference TS LCO 3.0.9.

The proposed changes are based on Technical Specification Task Force (TSTF) Traveler TSTF-372, "Addition of LCO 3.0.8, Inoperability of Snubbers," Revision 4 (TSTF-372)(ADAMS Accession No. ML041200567), which was approved generically for the Standard Technical Specifications (STSs) (NUREGs 1430–1434) by the U.S. Nuclear Regulatory Commission (NRC or Commission). The NRC staff published a notice of availability of this STS change in the *Federal Register* (FR) on May 4, 2005 (70 FR 23252) as part of the Consolidated Line Item Improvement Process. The notice included a model safety evaluation (SE) that may be referenced by licensees in plant-specific applications to adopt the TSTF-372 changes. In its application, the licensee stated that the justifications presented in TSTF-372 and the model SE are applicable to the facilities and justify the proposed TS changes. Since Braidwood and Byron each currently contain an LCO numbered 3.0.8, the new proposed LCO is numbered 3.0.9. This deviation is administrative since it does not change the current TS requirements or change the TSTF-372 LCO.

The SE that follows is based on the TSTF-372 model SE. Braidwood and Byron are both pressurized-water reactors (PWRs) located in the Central U.S. Therefore, the discussions in TSTF-372 and the associated model SE regarding boiling-water reactors (BWRs) and West Coast facilities are not applicable to Braidwood and Byron.

The TSTF-372 added LCO 3.0.8 (herein referred to LCO 3.0.9 to be consistent with the Braidwood and Byron specific request) to the STSs, which allows a delay time for entering a snubber supported system TS, when the inoperability is due solely to an inoperable snubber, if risk is assessed and managed. The postulated seismic event requiring snubbers is a low-probability occurrence, and the overall TS system safety function would still be available for the vast majority of anticipated challenges.

The TSTF-372 was developed as one of the industry's initiatives under the risk-informed TSs program. These initiatives are intended to maintain or improve safety through the incorporation of risk assessment and management techniques in the TSs, while reducing unnecessary burden and making TS requirements consistent with the Commission's other risk-informed regulatory requirements, in particular the Maintenance Rule (Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants").

Based on TSTF-372 the licensee's proposed change would add a new LCO (LCO 3.0.9) to the facilities' TS. LCO 3.0.9 would allow the licensee to delay declaring an LCO not met for equipment that is supported by snubbers unable to perform their associated support functions when the risk associated with the delay is assessed and managed. The licensee's proposed new LCO 3.0.9 states:

> When one or more required snubbers are unable to perform their associated support function(s), any affected supported LCO(s) are not required to be declared not met solely for this reason if risk is assessed and managed, and:
>
> a. the snubbers not able to perform their associated support function(s) are associated with only one train or subsystem of a multiple train or subsystem supported system or are associated with a single train or subsystem supported system and are able to perform their associated support function within 72 hours; or
>
> b. the snubbers not able to perform their associated support function(s) are associated with more than one train or subsystem of a multiple train or subsystem supported system and are able to perform their associated support function within 12 hours.
>
> At the end of the specified period the required snubbers must be able to perform their associated support function(s), or the affected supported system LCO(s) shall be declared not met.

Consistent with TSTF-372, a conforming change would also be made to LCO 3.0.1 to reference the new LCO 3.0.9. For Braidwood and Byron, TS LCO 3.0.1 currently states:

> LCOs shall be met during the MODES or other specified conditions in the Applicability, except as provided in LCO 3.0.2 and LCO 3.0.7.

LCO 3.0.1 would be revised to read as follows:

> LCOs shall be met during the MODES or other specified conditions in the Applicability, except as provided in LCO 3.0.2, LCO 3.0.7, and LCO 3.0.9.

2.0 <u>REGULATORY EVALUATION</u>

In 10 CFR 50.36, "Technical specifications," the Commission established its regulatory requirements related to the content of the TSs. Pursuant to 10 CFR 50.36, TSs are required to include items in the following five specific categories: (1) safety limits, limiting safety system settings, and limiting control settings; (2) LCOs; (3) surveillance requirements; (4) design features; and (5) administrative controls. The rule does not specify the particular requirements to be included in a plant's TSs. As stated in 10 CFR 50.36(c)(2)(i):

> Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met.

For Braidwood and Byron, TS Section 3.0, "Limiting Condition for Operation (LCO) Applicability," provides details or general application rules for complying with the LCOs.

Snubbers are chosen in lieu of rigid supports in areas where restricting thermal growth during normal operation would induce excessive stresses in the piping nozzles or other equipment. Although they are classified as component standard supports, they are not designed to provide any transmission of force during normal plant operations. However, in the presence of dynamic transient loadings, which are induced by seismic events as well as by plant accidents and transients, a snubber functions as a rigid support. The location and size of the snubbers are determined by stress analyses based on different combinations of load conditions, depending on the design classification of the particular piping.

Prior to the conversion to the improved STSs, TSs included requirements that applied directly to snubbers. These requirements included:

- A requirement that snubbers be operable and in service when the supported equipment is required to be operable;

- A requirement that snubber removal for testing be done only during plant shutdown;

- A requirement that snubber removal for testing be done on a one-at-a-time basis when supported equipment is required to be operable during shutdown;

- A requirement to repair or replace within 72 hours any snubbers, found to be inoperable during operation in Modes 1 through 4, to avoid declaring any supported equipment inoperable;

- A requirement that each snubber be demonstrated operable by periodic visual inspections; and
- A requirement to perform operability tests on a representative sample of at least 10 percent of plant snubbers, at least once every 18 months during shutdown.

In the late 1980s, a joint initiative of the NRC staff and industry was undertaken to improve the STSs. This effort identified the snubbers as candidates for relocation to a licensee-controlled document, based on the fact that the TS requirements for snubbers did not meet any of the four criteria in 10 CFR 50.36(c)(2)(ii) for inclusion in the improved STSs. The NRC approved the relocation without placing any restriction on the use of the relocated requirements. However, this relocation resulted in different interpretations between the NRC and the industry regarding its implementation.

The NRC has stated that, since snubbers support safety equipment included in the TSs, the definition of OPERABILITY must be used to immediately evaluate equipment supported by a removed snubber and, if found inoperable, the appropriate TS required actions must be entered. This interpretation has, in practice, eliminated the 72-hour delay to enter the actions for the supported equipment that existed prior to the conversion to the improved STSs (the only exception is if the supported system has been analyzed and determined to be OPERABLE without the snubber). The industry has argued that since the NRC approved the relocation without placing any restriction on the use of the relocated requirements, the licensee-controlled document requirements for snubbers should be invoked before the supported system's TS requirements become applicable. The industry's interpretation would, in effect, restore the 72-hour delay to enter the actions for the supported equipment that existed prior to the conversion to the improved STS. The industry's proposal would allow a time delay for all conditions, including snubber removal for testing at power.

The option to relocate the snubbers to a licensee-controlled document, as part of the conversion to improved STSs, has resulted in non-uniform and inconsistent treatment of snubbers. On the one hand, plants that have relocated snubbers from their TSs to licensee-controlled documents are allowed to change the requirements for snubbers under the auspices of 10 CFR 50.59, "Changes, tests, and experiments," provided the requirements of 10 CFR 50.55a, "Codes and standards," continue to be met, but they are not allowed a 72-hour delay before they enter the actions for the supported equipment. On the other hand, plants that have not converted to the improved STSs have retained the 72-hour delay if snubbers are found to be inoperable, but they are only allowed to change TS requirements for snubbers through a license amendment. A few plants that converted to the improved STSs chose not to relocate the snubbers to a licensee-controlled document and, thus, retained the 72-hour delay. In addition, it is important to note that plants that have relocated the snubber requirements can perform functional tests on the snubbers at power (as long as they enter the actions for the supported equipment), unlike plants that still have snubber requirements in TSs. Some potential undesirable consequences of this inconsistent treatment of snubbers are:

- Performance of testing during crowded time period windows when the supported system is inoperable with the potential to reduce the snubber testing to a minimum since the snubber requirements relocated from TSs are controlled by the licensee;

- Performance of testing during crowded windows when the supported system is inoperable with the potential to increase the unavailability of safety systems; and

- Performance of testing and maintenance on snubbers affecting multiple trains of the same supported system during the 7 hours allotted before entering MODE 3 under LCO 3.0.3.

To remove the inconsistency in the treatment of snubbers among plants, the TSTF proposed a risk-informed TS change that introduces a delay time before entering the actions for the supported equipment, when one or more snubbers are found inoperable or removed for testing, if risk is assessed and managed. Such a delay time will provide needed flexibility in the performance of maintenance and testing during power operation and at the same time will enhance overall plant safety by:

- Avoiding unnecessary unscheduled plant shutdowns and, thus, minimizing plant transition and realignment risks;

- Avoiding reduced snubber testing and, thus, increasing the availability of snubbers to perform their supporting function;

- Performing most of the required testing and maintenance during the delay time when the supported system is available to mitigate most challenges and, thus, avoiding increases in safety system unavailability; and

- Providing explicit risk-informed guidance in areas in which that guidance currently does not exist, such as the treatment of snubbers impacting more than one redundant train of a supported system.

3.0 TECHNICAL EVALUATION

The industry submitted TSTF-372 in support of the proposed TS change, which documents a risk-informed analysis of the proposed TS change. The NRC staff used the probabilistic risk assessment (PRA) results and insights from TSTF-372, in combination with deterministic and defense-in-depth arguments, to evaluate the proposed delay times for entering the actions for the supported equipment associated with inoperable snubbers at nuclear power plants. In the model SE for TSTF-372, the staff used the guidance in Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" dated July 1998, and RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," dated August 1998 (ADAMS Accession Nos. ML003740133 and ML003740176, respectively). However, for Exelon's application, the NRC staff considered the current guidance in RG 1.174, Revision 3, dated January 2018, and RG 1.177, Revision 1, dated May 2011 (ADAMS Accession Nos. ML17317A256 and ML100910008, respectively).

The risk impact associated with the proposed delay times for entering the TS actions for the supported equipment can be assessed using the same approach as for allowed completion time extensions. Therefore, the NRC evaluated the proposed change using the following three-tiered approach recommended in RG 1.177 for evaluating proposed extensions in currently allowed completion times:

- The first tier involves the assessment of the change in plant risk due to the proposed TS change, as expressed by the change in core damage frequency (ΔCDF), the change in large early release frequency (ΔLERF), the incremental conditional core damage probability (ICCDP), and the incremental conditional large early release probability (ICLERP). The assessed ΔCDF and ΔLERF values are compared to acceptance guidelines, consistent with the Commission's Safety Goal Policy Statement, as documented in RG 1.174, so that the plant's baseline risk is maintained within a minimal range. The assessed ICCDP and ICLERP values are compared to acceptance guidelines provided in RG 1.177, which aim at ensuring that the plant risk does not increase unacceptably during the period the equipment is taken out of service.

- The second tier involves the identification of potentially high-risk configurations that could exist if equipment in addition to that associated with the change were to be taken out of service simultaneously, or other risk-significant operational factors such as concurrent equipment testing were also involved. The objective is to ensure that appropriate restrictions are in place to avoid any potential high-risk configurations.

- The third tier involves the establishment of an overall configuration risk management program (CRMP) to ensure that potentially risk-significant configurations resulting from maintenance and other operational activities are identified. The objective of the CRMP is to manage configuration-specific risk by appropriate scheduling of plant activities and/or appropriate compensatory measures.

In TSTF-372, a simplified bounding risk assessment was performed to justify the proposed addition of LCO 3.0.9 to the TSs. This approach was necessitated by: (1) the general nature of the proposed TS changes (i.e., they apply to all plants and are associated with an undetermined number of snubbers that are not able to perform their function), (2) the lack of detailed engineering analyses that establish the relationship between earthquake level and supported system pipe failure probability when one or more snubbers are inoperable, and (3) the lack of seismic risk assessment models for most plants. The simplified risk assessment was based on the following major assumptions, which the NRC staff finds acceptable, as discussed below:

- The accident sequences contributing to the risk increase associated with the proposed TS changes are assumed to be initiated by a seismically-induced loss-of-offsite power (LOOP) event with concurrent loss of all safety system trains supported by the out-of-service snubbers. In the case of snubbers associated with more than one train (or subsystem) of the same system, it is assumed that all affected trains (or subsystems) of the supported system are failed. This assumption was introduced to allow the performance of a simple bounding risk assessment approach applicable to all plants. This approach was selected due to the lack of detailed plant-specific seismic risk assessments for most plants and the lack of fragility data for piping when one or more supporting snubbers are inoperable.

- The LOOP event is assumed to occur due to the seismically-induced failure of the ceramic insulators used in the power distribution systems. These ceramic insulators have a high confidence (95 percent) of low probability (5 percent) of failure (HCLPF) at a peak ground acceleration about 0.1g (0.1 times the acceleration due to gravity (g)). Thus, a magnitude 0.1g earthquake is conservatively assumed to have 5 percent probability of causing a LOOP initiating event. The fact that no LOOP events caused by higher magnitude earthquakes were considered is justified because: (1) the frequency of earthquakes decreases with increasing magnitude and (2) historical data (References 1 and 2) indicate that the mean seismic capacity of ceramic insulators (used in seismic PRAs), in terms of peak ground acceleration, is about 0.3g, which is significantly higher than the 0.1g HCLPF value. Therefore, the simplified analysis, even though it does not consider LOOP events caused by earthquakes of a magnitude higher than 0.1g, bounds a detailed analysis that would use mean seismic failure probabilities (fragilities) for the ceramic insulators.

- Analytical and experimental results obtained in the mid-1980s, as part of the industry's snubber reduction program (References 1 and 3), indicated that piping systems have large margins against seismic stress. The assumption that a magnitude 0.1g earthquake would cause the failure of all safety system trains supported by the out-of-service snubbers is very conservative, because safety piping systems could withstand much higher seismic stresses even when one or more supporting snubbers are out of service. The actual piping failure probability is a function of the allowable stress and the number of snubbers removed for maintenance or testing. Since the licensee-controlled testing is done on only a small representative sample (about 10 percent) of the total snubber population, typically only a few snubbers supporting a given safety system are out for testing at a time. Furthermore, since the testing of snubbers is a planned activity, licensees have flexibility in selecting a sample set of snubbers for testing from a much larger population by conducting configuration-specific engineering and/or risk assessments. Such a selection of snubbers for testing provides confidence that the supported systems would perform their functions in the presence of a design-basis earthquake and other dynamic loads, and, in any case, the risk impact of the activity will remain within the limits of acceptability defined in risk-informed RGs 1.174 and 1.177.

- The analysis assumed that one train (or subsystem) of <u>all</u> safety systems supported by snubbers is unavailable during snubber testing or maintenance (an entire system is assumed unavailable if a removed snubber is associated with both trains of a two-train system). This is a very conservative assumption for the case of corrective maintenance, since it is unlikely that a visual inspection will reveal that one or more snubbers across all supported systems are inoperable. This assumption is also conservative for the case of the licensee-controlled testing of snubbers, since such testing is performed only on a small representative sample.

- In general, the TSTF-372 risk assessment did not credit recovery actions or alternative means of performing a function, such as the function performed by a system assumed failed (e.g., when LCO 3.0.9.b applies). However, most plants have reliable alternative means of performing certain critical functions. For example, feed and bleed (F&B) can be used to remove heat in most pressurized water reactors (PWRs) when auxiliary feedwater (AFW), the most important system in mitigating LOOP accidents, is unavailable.. Credit for recovery actions to provide core cooling using alternative means could have been applied to most plants.

- The earthquake frequency at the 0.1g level was assumed to be 10^{-3}/year for Central and Eastern U.S. plants and 10^{-1}/year for West Coast plants. Each of these two values envelop the range of earthquake frequency values at the 0.1g level, for the Central and Eastern U.S. plants and the West Coast plants, respectively (References 2 and 4).

- The risk impact associated with non-LOOP accident sequences (e.g., seismically initiated loss-of-coolant accident (LOCA) or anticipated transient without scram) was not assessed. However, this risk impact is small compared to the risk impact associated with the LOOP accident sequences modeled in the simplified bounding risk assessment. Non-LOOP accident sequences, due to the ruggedness of nuclear power plant designs, require seismically-induced failures that occur at earthquake levels above 0.3g. Thus, the frequency of earthquakes initiating non-LOOP accident sequences is much smaller than the frequency of seismically-initiated LOOP events. Furthermore, because of the conservative assumption made for LOOP sequences that a 0.1g level earthquake would fail all piping associated with inoperable snubbers, non-LOOP sequences would not include any more failures associated with inoperable snubbers than would LOOP sequences. Therefore, the risk impact of inoperable snubbers associated with non-LOOP accident sequences is small compared to the risk impact associated with the LOOP accident sequences modeled in the simplified bounding risk assessment.

- The risk impact of dynamic loadings other than seismic loads was not assessed. These shock-type loads include thrust loads, blowdown loads, waterhammer loads, steamhammer loads, LOCA loads, and pipe rupture loads. However, there are some important distinctions between nonseismic (shock-type) loads and seismic loads, which indicate that, in general, the risk impact of the out-of-service snubbers is smaller for nonseismic loads than for seismic loads. First, while a seismic load affects the entire plant, the impact of a nonseismic load is localized to a certain system or area of the plant. Second, although nonseismic shock loads may be higher in total force and the impact could be as much or more than seismic loads, generally they are of much shorter duration than seismic loads. Third, the impact of nonseismic loads is more plant specific, and, thus, is harder to analyze generically than is the impact of seismic loads. For these reasons, licensees will be required to confirm, every time LCO 3.0.9.a is used, that at least one train of each system that is supported by the inoperable snubber(s) would remain capable of performing the system's required safety or support functions for postulated design loads other than seismic loads.

3.1 Risk Assessment Results and Insights

The results and insights from the NRC staff's evaluation using the three-tiered approach in RG 1.177 to support the review of the proposed addition of LCO 3.0.9 to the TSs are summarized and evaluated in Sections 3.1.1 to 3.1.3 below.

3.1.1 Risk Impact

The bounding risk assessment approach, discussed in Section 3.0, was applied generically for all U.S. operating nuclear power plants. Risk assessments were performed by the NRC staff for two categories of plants, Central and Eastern U.S. plants and West Coast plants, based on historical seismic hazard curves (earthquake frequencies and associated magnitudes). The Central and Eastern U.S. category includes the vast majority of the U.S. nuclear power plants (Reference 4), including the licensee's facilities. The risk assessment for the West Coast plants is not discussed in this SE because it is not applicable to the licensee's facilities. For each category of plants, two risk assessments were performed by the staff:

- The first risk assessment applies to cases where all inoperable snubbers are associated with only one train (or subsystem) of the impacted safety systems. It was conservatively assumed that a single train (or subsystem) of each safety system is unavailable. It was also assumed that the probability of non-mitigation using the unaffected redundant trains (or subsystems) is 2 percent. This is a conservative value given that for core damage to occur under these conditions two or more failures are required.

- The second risk assessment applies to the case where one or more of the inoperable snubbers are associated with multiple trains (or subsystems) of the same safety systems. For the Central and Eastern U.S. plants, it was assumed in this bounding analysis that all safety systems supported by snubbers are unavailable to mitigate the accident. However, credit for recovery actions to provide core cooling using alternative means was applied to the Central and Eastern U.S. plants.

The results of the risk assessments for Central and Eastern U.S. plants, in terms of core damage and large early release risk impacts, are summarized in Table 1. The first row lists the conditional risk increase (ΔR_{CDF}), in terms of core damage frequency (CDF), caused by the out-of-service snubbers (as assumed in the bounding analysis). The second and third rows list the ICCDP and the ICLERP values, respectively. For the case where all inoperable snubbers are associated with only one train (or subsystem) of the supported safety systems, the ICCDP was obtained by multiplying the corresponding ΔR_{CDF} value by the time fraction of the proposed 72-hour delay to enter the actions for the supported equipment. For the case where one or more of the inoperable snubbers are associated with multiple trains (or subsystems) of the same safety system, the ICCDP was obtained by multiplying the corresponding ΔR_{CDF} value by the time fraction of the proposed 12-hour delay to enter the actions for the supported equipment. The ICLERP values were obtained by multiplying the corresponding ICCDP values by 0.1 (i.e., by assuming that the ICLERP value is an order of magnitude less than the ICCDP). This assumption is conservative because containment bypass scenarios, such as interfacing system LOCAs, would not be uniquely affected by the out-of-service snubbers.

Finally, the fourth and fifth rows list the assessed ΔCDF and ΔLERF values, respectively. These values were obtained by dividing the corresponding ICCDP and ICLERP values by 1.5 (i.e., by assuming that the snubbers are tested every 18 months, as was the case before the snubbers were relocated to a licensee-controlled document). This assumption is reasonable because: (1) it is not expected that licensees would test the snubbers more often than what used to be required by the TSs, and (2) testing of snubbers is associated with higher risk impact than is the average corrective maintenance of snubbers found inoperable by visual inspection (testing is expected to involve significantly more snubbers out of service than corrective maintenance). The assessed ΔCDF and ΔLERF values are compared to acceptance guidelines, consistent with the Commission's Safety Goal Policy Statement, as documented in RG 1.174, so that the plant's baseline risk is maintained within a minimal range. This comparison indicates that the addition of LCO 3.0.9 to the existing TSs would have an insignificant risk impact.

Table 1: Bounding Risk Assessment Results for Central and Eastern U.S. Plants for Snubbers Impacting a Single Train and Multiple Trains of a Supported System

	Single Train	Multiple Train
ΔR_{CDF}/year	1×10^{-6}	5×10^{-6}
ICCDP	8×10^{-9}	7×10^{-9}
ICLERP	8×10^{-10}	7×10^{-10}
ΔCDF/year	5×10^{-9}	5×10^{-9}
ΔLERF/year	5×10^{-10}	5×10^{-10}

The assessed ΔCDF and ΔLERF values meet the acceptance criteria of 10^{-6}/year and 10^{-7}/year, respectively, based on guidance provided in RG 1.174, Revision 3. This conclusion is true without taking any credit for the removal of potential undesirable consequences associated with the current inconsistent treatment of snubbers (e.g., reduced snubber testing frequency, increased safety system unavailability, and treatment of snubbers impacting multiple trains) discussed in Section 2.0 above, and given the bounding nature of the risk assessment.

The assessed ICCDP and ICLERP values are compared to acceptance guidelines provided in RG 1.177, Revision 1, which aim at ensuring that the plant risk does not increase unacceptably during the period the equipment is taken out of service. This comparison indicates that the addition of LCO 3.0.9 to the existing TSs meets the RG 1.177, Revision 1, numerical guidelines of 10^{-6} for ICCDP and 10^{-7} for ICLERP.

The risk assessment results of Table 1 are also compared to NRC-endorsed guidance in Section 11 of NUMARC 93-01, Revision 4F (Reference 5), for implementing the requirements of paragraph (a)(4) of the Maintenance Rule (10 CFR 50.65). This guidance is summarized in Table 2. Guidance regarding the acceptability of conditional risk increase in terms of CDF (i.e., ΔR_{CDF}) for a planned configuration is provided. The NUMARC 93-01 guidance states that a specific configuration that is associated with a CDF higher than 10^{-3}/year should be carefully considered before entering voluntarily. However, since the assessed conditional risk increase, ΔR_{CDF}, is significantly less than 10^{-3}/year, plant configurations including out-of-service snubbers and other equipment may be entered voluntarily if supported by the results of the risk assessment required by 10 CFR 50.65(a)(4), by LCO 3.0.9, or by other TSs.

Table 2: Guidance for Implementing 10 CFR 50.65(a)(4)

ΔR_{CDF}	**Guidance**	
Greater than 10^{-3}/year	Configuration should be carefully considered before entering voluntarily.	
ICCDP	**Guidance**	**ICLERP**
Greater than 10^{-5}	Configuration should not normally be entered voluntarily.	Greater than 10^{-6}
10^{-6} to 10^{-5}	Assess non-quantifiable factors. Establish risk management actions.	10^{-7} to 10^{-6}
Less than 10^{-6}	Normal work controls.	Less than 10^{-7}

Guidance regarding the acceptability of ICCDP and ICLERP values for a specific planned configuration and the establishment of risk management actions is also provided in NUMARC 93-01. This guidance, as shown in Table 2, states that a specific plant configuration with ICCDP and ICLERP values less than 10^{-6} and 10^{-7}, respectively, can be entered with normal work controls. Table 1 shows that for all Central and Eastern U.S. plants the conservatively assessed ICCDP and ICLERP values are over an order of magnitude less than what is recommended as the threshold for the normal work controls region. Thus, the risk contribution from out-of-service snubbers is within the normal range of maintenance activities carried out at Central and Eastern U.S. plants. Therefore, plant configurations involving out-of-service snubbers and other equipment may be entered voluntarily if supported by the results of the risk assessment required by 10 CFR 50.65(a)(4), by LCO 3.0.9, or by other TSs.

The NRC staff finds that the risk assessment results support the proposed addition of LCO 3.0.9 to the TSs. The risk increases associated with this TS change will be insignificant (based on guidance provided in RGs 1.174 and 1.177) and within the range of risks associated with normal maintenance activities. In addition, LCO 3.0.9 will remove potential undesirable consequences stemming from the current inconsistent treatment of snubbers in the TSs, such as reduced frequency of snubber testing, increased safety system unavailability, and the treatment of snubbers impacting multiple trains.

3.1.2 Identification of High-Risk Configurations

The second tier of the three-tiered approach recommended in RG 1.177, Revision 1, involves the identification of potentially high-risk configurations that could exist if equipment, in addition to that associated with the TS change, were to be taken out of service simultaneously. Insights from the risk assessments, in conjunction with important assumptions made in the analysis and defense-in-depth considerations, were used to identify such configurations. To avoid these potentially high-risk configurations, specific restrictions to the implementation of the proposed TS changes were identified in the model SE for TSTF-372.

For cases where all inoperable snubbers are associated with only one train (or subsystem) of the impacted systems (i.e., when LCO 3.0.9.a applies), it was assumed in the analysis that there will be unaffected redundant trains (or subsystems) available to mitigate the seismically-initiated LOOP accident sequences. This assumption implies that there will be at least one success path available when LCO 3.0.9.a applies. Therefore, potentially high-risk configurations can be avoided by ensuring that such a success path exists when LCO 3.0.9.a applies. Based on a review of the accident sequences that contribute to the risk increase associated with LCO 3.0.9.a, as modeled by the simplified bounding analysis (i.e., accident sequences initiated by a seismically-induced LOOP event with concurrent loss of all safety system trains supported by the out-of-service snubbers), the following restriction specified in the model SE for TSTF-372 is applicable to Braidwood and Byron to prevent potentially high-risk configurations:

- For the facilities (PWR plants), at least one AFW [auxiliary feedwater] train (including a minimum set of supporting equipment required for its successful operation) not associated with the inoperable snubber(s), must be available when LCO 3.0.9a is used.

For cases where one or more of the inoperable snubbers are associated with multiple trains (or subsystems) of the same safety system (i.e., when LCO 3.0.9.b applies), it was assumed in the TSTF-372 bounding analysis for Central and Eastern U.S. plants that all safety systems are unavailable to mitigate the accident. Based on a review of the accident sequences that contribute to the risk increase associated with LCO 3.0.9.b (as modeled by the simplified bounding analysis) and on defense-in-depth considerations, the following restriction specified in the model SE for TSTF-372 is applicable to the Braidwood and Byron to prevent potentially high-risk configurations:

- When LCO 3.0.9b is used at PWR plants, at least one AFW train (including a minimum set of supporting equipment required for its successful operation) not associated with the inoperable snubber(s), or some alternative means of core cooling (e.g., F&B [feed and bleed], firewater system or "aggressive secondary cooldown" using the steam generators) must be available.

3.1.3 Configuration Risk Management

The third tier of the three-tiered approach recommended in RG 1.177, Revision 1, involves the establishment of an overall CRMP to ensure that potentially risk-significant configurations resulting from maintenance and other operational activities are identified. The objective of the CRMP is to manage configuration-specific risk by appropriate scheduling of plant activities and/or appropriate compensatory measures. This objective is met by licensee programs to comply with the requirements of paragraph (a)(4) of the Maintenance Rule (10 CFR 50.65) to assess and manage risk resulting from maintenance activities, and by the TSs requiring risk assessments and management using 10 CFR 50.65(a)(4) processes if no maintenance is in progress. These programs can support licensee decisionmaking regarding the appropriate actions to manage risk whenever a risk-informed TS is entered. Since the 10 CFR 50.65(a)(4) guidance in Section 11 of NUMARC 93-01, Revision 4F, does not address seismic risk, licensees adopting this change must ensure that use of the proposed LCO 3.0.9 is considered with respect to other plant maintenance activities and integrated into the existing 10 CFR 50.65(a)(4) process, whether the process is invoked by a TS or by 10 CFR 50.65(a)(4) itself.

3.1.4 Optional Changes and Variations

The licensee did not propose any technical variations from the TS changes described in TSTF-372, Revision 4, or the model SE.

Since Braidwood and Byron each currently contain an LCO numbered 3.0.8, the new proposed LCO is numbered 3.0.9. This deviation is administrative since it does not change the current TS requirements or change the TSTF-372 LCO.

3.2 Summary and Conclusions

The option to relocate the snubbers to a licensee-controlled document, as part of the conversion to improved STSs, has resulted in non-uniform and inconsistent treatment of snubbers. Some potential undesirable consequences of this inconsistent treatment of snubbers are:

- Performance of testing during crowded windows when the supported system is inoperable, with the potential to reduce the snubber testing to a minimum (within the requirements of 10 CFR 50.55a) since the relocated snubber requirements are controlled by the licensee;

- Performance of testing during crowded windows when the supported system is inoperable with the potential to increase the unavailability of safety systems; or

- Performance of testing and maintenance on snubbers affecting multiple trains of the same supported system during the 7 hours allotted before entering MODE 3 under LCO 3.0.3.

To remove the inconsistency among plants in the treatment of snubbers, the industry proposed a risk-informed TS change that introduces a delay time before entering the actions for the supported equipment when one or more snubbers are found inoperable or removed for testing. The delay time will provide needed flexibility in the performance of maintenance and testing during power operation and, at the same time, will enhance overall plant safety by: (1) avoiding unnecessary unscheduled plant shutdowns, thus, minimizing plant transition and realignment risks; (2) avoiding reduced snubber testing, thus, increasing the availability of snubbers to perform their supporting function; (3) performing most of the required testing and maintenance during the delay time when the supported system is available to mitigate most challenges, thus avoiding increases in safety system unavailability; and (4) providing explicit risk-informed guidance in areas where that guidance currently does not exist, such as the treatment of snubbers impacting more than one redundant train of a supported system.

The risk impact of the proposed TS changes was assessed generically following the three-tiered approach recommended in RG 1.177, Revision 1. A simplified bounding risk assessment was performed to justify the proposed TS changes. This bounding assessment assumed that the risk increase associated with the proposed addition of LCO 3.0.9 to the TSs is associated with accident sequences initiated by a seismically-induced LOOP event with concurrent loss of all safety system trains supported by the out-of-service snubbers. In the case of snubbers associated with more than one train, it is assumed that all affected trains of the supported system are failed. This assumption was introduced to allow the performance of a simple bounding risk assessment approach with application to all plants and was selected due to the lack of detailed plant-specific seismic risk assessments for most plants and the lack of fragility data for piping when one or more supporting snubbers are inoperable. The impact from the addition of the proposed LCO 3.0.9 to the TSs on defense-in-depth was also evaluated in conjunction with the risk assessment results.

Based on this integrated evaluation, the NRC staff concludes that the proposed addition of LCO 3.0.9 to the licensee's TSs would lead to insignificant risk increases, if any. Indeed, this conclusion is true without taking any credit for the removal of potential undesirable consequences associated with the current inconsistent treatment of snubbers, such as the effects of avoiding a potential reduction in the snubber testing frequency and increased safety system unavailability.

Consistent with the NRC staff's approval of TSTF-372, Revision 4, as documented in the model SE, and inherent in the implementation of TSTF-372, the licensees must operate in accordance with the following stipulations applicable to PWRs located in the Central and Eastern U.S.:

1. Appropriate plant procedures and administrative controls will be used to implement the following Tier 2 Restrictions:

    a. At least one AFW train (including a minimum set of supporting equipment required for its successful operation) not associated with the inoperable snubber(s), must be available when LCO 3.0.9a is used.

    b. At least one AFW train (including a minimum set of supporting equipment required for its successful operation) not associated with the inoperable snubber(s), or some alternative means of core cooling (e.g., F&B [feed & bleed], fire water system or "aggressive secondary cooldown" using the steam generators) must be available when LCO 3.0.9b is used.

    c. Every time the provisions of LCO 3.0.9 are used, licensees will be required to confirm that at least one train (or subsystem) of systems supported by the inoperable snubbers would remain capable of performing the system's required safety or support functions for postulated design loads other than seismic loads. LCO 3.0.9 does not apply to nonseismic snubbers. In addition, a record of the design function of the inoperable snubber (i.e., seismic vs. non-seismic), the implementation of any applicable Tier 2 restrictions, and the associated plant configuration shall all be available on a recoverable basis for staff inspection.

2. When the licensee implements the provisions of LCO 3.0.9 for snubbers, which include delay times to enter the actions for the supported equipment when one or more snubbers are out of service for maintenance or testing, it must be done in accordance with an overall CRMP to ensure that potentially risk-significant configurations resulting from maintenance and other operational activities are identified and avoided, as discussed in the proposed TS Bases. This objective is met by licensee programs to comply with the requirements of paragraph (a)(4) of the Maintenance Rule (10 CFR 50.65) to assess and manage risk resulting from maintenance activities or when this process is invoked by LCO 3.0.9 or other TSs. These programs can support licensee decisionmaking regarding the appropriate actions to manage risk whenever a risk-informed TS is entered. Since the 10 CFR 50.65(a)(4) guidance in Section 11 of NUMARC 93-01 does not address seismic risk, licensees adopting this change must ensure that the proposed LCO 3.0.9 is considered in conjunction with other plant maintenance activities and integrated into the existing 10 CFR 50.65(a)(4) process. In

the absence of a detailed seismic PRA, a bounding risk assessment, such as that used in this SE, shall be followed.

The licensee's January 31, 2019, letter acknowledged that the stipulations above are applicable to Braidwood and Byron, and that it will operate in accordance with these stipulations. The letter also provided proposed TS Bases for LCO 3.0.9 which included language consistent with these stipulations; however, in accordance with 10 CFR 50.36(a)(1), the TS bases shall not become part of the TSs. Therefore, the NRC staff did not make a finding regarding the acceptability of the TS bases changes.

In addition, the licensee's January 31, 2019, letter provided a regulatory commitment. The licensee committed to provide guidance and details on how to implement the above stipulations in the TS Bases for LCO 3.0.9 for the facilities. The NRC staff finds these commitments to be acceptable, but did not rely on them to form the basis for its conclusions in this SE.

Based on its review as documented above, the NRC staff finds that the proposed TS changes are acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Illinois State official was notified of the proposed issuance of the amendments on July 8, 2019. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (84 FR 19970, dated May 7, 2019). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

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

7.0 REFERENCES

1. Budnitz, R. J., et al., "An Approach to the Quantification of Seismic Margins in Nuclear Power Plants," NUREG/CR-4334, Lawrence Livermore National Laboratory, July 1985.

Available for examination and purchase at the NRC's Public Document Room, Room O1-F21,One White Flint North, 11555 Rockville Pile, Rockville, Maryland 20852.

2. Advanced Light Water Reactor Utility Requirements Document, Volume 2, ALWR Evolutionary Plant, PRA Key Assumptions and Groundrules, Electric Power Research Institute, August 1990. Available from Electric Power Research Institute, 3420 Hillview Avenue, Palo Alto, CA 94304. Telephone: 650-855-2000.

3. Bier V. M., et al., "Development and Application of a Comprehensive Framework for Assessing Alternative Approaches to Snubber Reduction," International Topical Conference on Probabilistic Safety Assessment and Risk Management PSA '87, Swiss Federal Institute of Technology, Zurich, August 30–September 4, 1987. Available from Swiss Federal Institute of Technology, Rämistrasse 101, Zürich, Canton of Zürich 8092

4. NUREG-1488, "Revised Livermore Seismic Hazard Estimates for Sixty-Nine Nuclear Power Plant Sites East of the Rocky Mountains," April 1994. Available for examination and purchase at the NRC's Public Document Room, Room O1-F21,One White Flint North, 11555 Rockville Pile, Rockville, Maryland 20852.

5. Nuclear Energy Institute, NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 4F, dated April 2018 (ADAMS Accession No. ML18120A069), as endorsed by NRC RG 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 4, dated August 2018 (ADAMS Accession No. ML18220B281).

Principal Contributor: C. Tilton, NRR

Date of Issuance: August 28, 2019

SUBJECT: BRAIDWOOD STATION, UNITS 1 AND 2, AND BYRON STATION, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENTS NOS. 202 and 208 RE: LIMITING CONDITION OF OPERATION FOR INOPERABILITY OF SNUBBERS (EPID L-2019-LLA-0023) DATED AUGUST 28, 2019

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