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

March 10, 2021

Mr. David P. Rhoades
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 - ISSUANCE OF AMENDMENT NOS. 220 AND 220 RE: A ONE-TIME DEFERRAL OF STEAM GENERATOR TUBE INSPECTIONS (EPID L-2020-LLA-0272 [COVID-19])

Dear Mr. Rhoades:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 220 to Renewed Facility Operating License No. NPF-72 and Amendment No. 220 to Renewed Facility Operating License No. NPF-77 for the Braidwood Station, Units 1 and 2, respectively. The amendments are in response to your application dated December 16, 2020, as supplemented by letter dated February 9, 2021.

The amendments revise Technical Specification (TS) 5.5.9, "Steam Generator (SG) Program," to allow a one-time deferral of the SG tube inspections from the spring of 2021 to the fall of 2022 refueling outage. The proposed change was requested in response to social distancing recommendations provided by the Centers for Disease Control and Prevention, which have been issued as a defensive measure against the spread of the Coronavirus Disease 2019. The Unit 2 amendment number is incremented because Unit 1 and Unit 2 have a common TS page.

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

Sincerely,

/RA/

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 and STN 50-457

Enclosures:
1. Amendment No. 220 to NPF-72
2. Amendment No. 220 to NPF-77
3. 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. 220
Renewed License No. NPF-72

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

   A. The application for amendment by Exelon Generation Company, LLC (the licensee) dated December 16, 2020, as supplemented by letter dated February 9, 2021, 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. 220 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 5 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Nancy L. Salgado

Digitally signed by Nancy L. Salgado
Date: 2021.03.10 07:54:11 -05'00'

Nancy L. Salgado, Chief
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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

Date of Issuance: March 10, 2021

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. 220
Renewed License No. NPF-77

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

   A. The application for amendment by Exelon Generation Company, LLC (the licensee) dated December 16, 2020, as supplemented by letter dated February 9, 2021, 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. 220 and the Environmental Protection Plan contained in Appendix B, both of which are attached to Renewed License No. NPF-77, 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 5 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Nancy L. Salgado

Digitally signed by Nancy L. Salgado
Date: 2021.03.10 07:54:39 -05'00'

Nancy L. Salgado, Chief
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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

Date of Issuance: March 10, 2021

ATTACHMENT TO LICENSE AMENDMENT NOS. 220 AND 220

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.

Renewed Facility Operating Licenses

REMOVE	INSERT
License NPF-72	License NPF-72
-3-	-3-
License NPF-77	License NPF-77
-3-	-3-

Technical Specifications

REMOVE	INSERT
5.5 – 9	5.5 – 9

(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) <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 as revised through Amendment No. 220 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. 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) Maximum Power Level

The licensee is authorized to operate the facility at reactor core power levels is 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. 220 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.

5.5.9 Steam Generator (SG) Program (continued)

The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. A degradation assessment shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG installation.

2. For Unit 1, after the first refueling outage following SG installation, inspect each SG at least every 72 effective full power months or at least every third refueling outage (whichever results in more frequent inspections), with the exception that each SG is to be inspected during the fourth refueling outage in A1R23 following inspections completed in refueling outage A1R19. In addition, the minimum number of tubes inspected at each scheduled inspection shall be the number of tubes in all SGs divided by the number of SG inspection outages scheduled in each inspection period as defined in a, b, c, and d below. If a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at this location and that may satisfy the applicable tube plugging criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated. The fraction of locations to be inspected for this potential type of degradation at this location at the end of the inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the inspection period after the determination that a new form of degradation could potentially be occurring at this location divided by the total number of times the SG is scheduled to be inspected in the inspection period. Each inspection period defined below may be extended up to 3 effective full power months to include a SG inspection outage in an inspection period and the subsequent inspection period begins at the conclusion of the included SG inspection outage.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 220

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-72 AND AMENDMENT NO. 220

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-77

EXELON GENERATION COMPANY, LLC

BRAIDWOOD STATION, UNITS 1 AND 2

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

1.0 INTRODUCTION

By letter dated December 16, 2020 (Reference 1), as supplemented by letter dated February 9, 2021 (Reference 2), Exelon Generation Company, LLC (the licensee), submitted a license amendment request (LAR) that proposed a one-time change to the technical specification (TS) for Braidwood Station, Unit 1 (Braidwood Unit 1).

The proposed changes would allow a one-time deferral of the SG tube inspections required in Section 5.5.9, "Steam Generator (SG) Program," of the Braidwood Unit 1 TS (Reference 3). The amendment number for Braidwood, Unit 2, would be incremented because Unit 1 and Unit 2 have common TS pages. The proposed changes were submitted in response to social distancing recommendations from the United States Centers for Disease Control and Prevention, which have been issued as a defensive measure against the spread of the Coronavirus Disease 2019 (COVID-19). The licensee requested that the changes be approved as a license amendment in accordance with Section 50.90, "Application for amendment of license, construction permit, or early site permit," of Title 10 of the *Code of Federal Regulations* (10 CFR), "Energy."

The February 9, 2021, supplement, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC) staff's original proposed finding of no significant hazards consideration published in the *Federal Register* on January 26, 2021 (86 FR 7116).

2.0 REGULATORY EVALUATION

2.1 Description of System

The SG tubes function as an integral part of the reactor coolant pressure boundary (RCPB) and, in addition, serve to isolate radiological fission products in the primary coolant from the secondary coolant and the environment. For the purposes of this safety evaluation, SG tube integrity means that the tubes can perform this safety function in accordance with the plant design and licensing basis.

2.2 Regulatory Requirements and Guidance

Fundamental regulatory requirements with respect to the integrity of the SG tubing are established in 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities." Specifically, General Design Criterion (GDC) 14, "Reactor coolant pressure boundary," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, states that the RCPB shall be "designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture." GDC 15, Reactor coolant system design," states that the reactor coolant system and associated auxiliary, control, and protection systems "shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences." GDC 30, "Quality of reactor coolant pressure boundary," states, in part, that components which are part of the RCPB shall be "designed, fabricated, erected, and tested to the highest quality standards practical." GDC 31, "Fracture prevention of reactor coolant pressure boundary," states, in part, that the RCPB "shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized." GDC 32, "Inspection of reactor coolant pressure boundary," states, in part, that RCPB components shall be designed to permit "periodic inspection and testing of important areas and features to assess their structural and leaktight integrity."

Braidwood Unit 1 received its construction permit on December 31, 1975. In Section 3.1, "Conformance with NRC General Design Criteria," of their Updated Final Safety Analysis Report (UFSAR) (Reference 4), the licensee stated that the Braidwood Stations "…fully satisfy and are in compliance with the NRC General Design Criteria."

Section 182(a) of the Atomic Energy Act requires nuclear power plant operating licenses to include TS as part of any license. In 10 CFR 50.36, "Technical specifications," the NRC's regulatory requirements related to the content of the TSs are established. The TSs for all current pressurized-water reactor licenses require that an SG Program be established and implemented to ensure that SG tube integrity is maintained.

2.3 Steam Generator Tube Integrity Requirements in the Braidwood Unit 1 Technical Specifications

At Braidwood, programs established by the licensee, including the SG Program, are listed in the administrative controls section of the TS to operate the facility in a safe manner. For Braidwood Unit 1, the requirements for performing SG tube inspections and repair are in TS Section 5.5.9, while the requirements for reporting the SG tube inspections and repair are in TS Section 5.6.9, "Steam Generator (SG) Tube Inspection Report."

For Braidwood Unit 1, SG tube integrity is maintained by meeting the performance criteria specified in TS Section 5.5.9.b for structural and leakage integrity. TS Section 5.5.9.a requires that a condition monitoring assessment be performed during each outage in which the SG tubes are inspected to confirm that the performance criteria are being met. TS Section 5.5.9.d includes provisions regarding the scope, frequency, and methods of SG tube inspections. These provisions require that the inspections be performed with the objective of detecting flaws of any type that may be present along the length of a tube and that may satisfy the applicable tube plugging criteria. The applicable tube plugging criterion specified in TS Section 5.5.9.c is that tubes found during in-service inspection to contain flaws with a depth equal to or exceeding 40 percent of the nominal wall thickness shall be plugged.

Braidwood Unit 1 TS Section 3.4.13, “RCS [Reactor Coolant System] Operational LEAKAGE,” includes a limit on operational primary-to-secondary leakage beyond which the plant must be promptly shut down. Should a flaw exceeding the tube plugging limit not be detected during the periodic tube surveillance required by the plant TS, the operational leakage limit provides added measures for timely plant shutdown before tube structural and leakage integrity are impaired, consistent with the design and licensing bases.

The Braidwood UFSAR (Reference 5) describes the analysis of the consequences of postulated design-basis accidents for steam line failures (Sections 15.1.5 and 15.1.6) and a SG tube rupture (Section 15.6.3). These analyses consider primary-to-secondary leakage that may occur during these events and must show that the radiological consequences do not exceed the applicable limits of 10 CFR 50.67, “Accident source term,” or 10 CFR 100.11, “Determination of exclusion area, low population zone, and population center distance,” for offsite doses; GDC 19, “Control room,” of 10 CFR Part 50, Appendix A, for control room operator doses (or some fraction thereof, as appropriate to the accident); or the NRC-approved licensing basis (e.g., a small fraction of these limits). No accident analyses for Braidwood Unit 1 are being changed because of the proposed amendment; thus, no radiological consequences of any accident analysis are being changed. The proposed changes maintain the accident analyses and consequences that the NRC has reviewed and approved for the postulated design-basis accidents for SG tubes.

3.0 TECHNICAL EVALUATION

3.1 Background

3.1.1 SG Design

Braidwood Unit 1 has four replacement SGs designed and fabricated by Babcock and Wilcox of Cambridge, Ontario, Canada, that were installed in 1998. Each SG has 6,633 thermally treated Alloy 690 (Alloy 690TT) tubes that have an outside diameter of 0.6875 inches and a nominal wall thickness of 0.040 inches. The tubes are supported by Type 410 stainless steel, lattice grid tube straight leg supports, and flat fan bar U-bend supports. The tubes were hydraulically expanded at each end for the full depth of the tubesheet. To reduce residual stress, the U-bend Rows 1-21 were stress-relieved after bending.

3.1.2 Operating Experience

All four SGs in Braidwood Unit 1 were last inspected in spring 2012 (Refueling Outage 16 (A1R16)) and fall 2016 (Refueling Outage 19 ((A1R19)). Full-length bobbin probe examinations of 100 percent of the in-service tubes in all four SGs were performed during A1R16 and A1R19. Additional information regarding the SG inspections at Braidwood Unit 1 is available in the spring 2012 and fall 2016 SG Tube Inspection Reports (References 6 and 7, respectively).

Since the SGs were placed in service in 1998, a total of 89 tubes have been plugged (32 in SG 1A, 40 in SG 1B, 16 in SG 1C, and 1 in SG 1D). Table 2, “Braidwood Unit 1 Plugging

History by Degradation Mechanism," of Attachment 1 to the licensee's LAR dated December 16, 2020, identifies the causes for tube plugging and the corresponding number of tubes plugged.

Braidwood Unit 1 has the following existing degradation mechanisms: mechanical wear at fan bar and lattice grid supports, and wear from foreign objects (FOs). Inspections during A1R19 reported a total of 108 wear indications from all mechanisms in the four Braidwood Unit 1 SGs. Table 1 summarizes the wear indications by mechanism reported during A1R19, the latest inspection.

Table 1: Wear Indications Reported in Braidwood Unit 1 SGs (A1R19, Fall 2016)

Wear Mechanism	**Total Number of Indications in Each SG**				
	SG 1A	**SG 1B**	**SG 1C**	**SG 1D**	**Total**
Fan Bar	8	18	38	20	84
Lattice Grid	4	4	2	1	11
FOs (New and Historical)	9	1	3	0	13

Reference 6 summarizes the secondary side inspections performed in spring 2012 (A1R16), which included a visual inspection of the top of tubesheet (TTS) and foreign object search and retrieval (FOSAR) after sludge lancing in all four SGs. In addition, the feedwater feedring, steam drum, and upper tube bundle in SG 1C were visually inspected during the A1R16 secondary side inspections. During the NRC staff's review (Reference 8) of Reference 7, the licensee clarified that no anomalies or degradation were found during the secondary side visual inspections.

Reference 7 summarizes the secondary side inspections performed in fall 2016 (A1R19), which included sludge lancing in all four SGs followed by a post sludge lance FOSAR. In addition, visual inspections of the feedwater feedring, steam drum, and upper tube bundle in SG 1D were performed during the A1R19 secondary side inspections. The licensee reported that no degradation or structural distortion was found during the secondary side visual inspections. Thirteen total indications of FO wear on twelve tubes were identified during A1R19. Three tubes with four FO wear indications and one tube containing only a possible loose part signal (no wear) were preventatively plugged and stabilized during A1R19 because the FO (weld slag) could not be removed during FOSAR.

The licensee stated that no primary-to-secondary leakage was noted during the past three operating cycles (20, 21, and 22). The licensee explained this further by stating that all trends are below 3 gallons per day (gpd), except for leak rate spikes due to multiple failures of a radiation monitor, a radiation monitor temporarily taken out of service, and air in-leakage issues. Section 3.2.1.a and Figure 1 of Attachment 1 to the LAR provides additional information regarding the leak rate spikes during the past three operating cycles.

3.2 <u>Proposed TS Changes</u>

3.2.1 <u>Current TS Requirements</u>

TS Section 5.5.9.d for Braidwood requires periodic SG tube inspections to be performed and specifies provisions to be met for such inspections. TS Section 5.5.9.d.1 specifies the tube inspection scope required to be met during the first refueling outage following installation. For Braidwood Unit 1, TS Section 5.5.9.d.2 states, "…after the first refueling outage following SG installation, inspect each SG at least every 72 effective full power months or at least every third refueling outage (whichever results in more frequent inspections). In addition, the minimum number of tubes inspected at each scheduled inspection shall be the number of tubes in all SGs divided by the number of SG inspection outages scheduled in each inspection period as defined in a, b, c, and d below." Additionally, 100 percent of the tubes are required to be inspected during each sequential period of 144 effective full power months (EFPM) (TS Section 5.5.9.d.2.a), 120 EFPM (TS Section 5.5.9.2.b), 96 EFPM (TS Section 5.5.9.2.c), and 72 EFPM (TS Section 5.5.9.2.d). Braidwood Unit 1 is currently in the second inspection period.

3.2.2 Description of Proposed TS Changes

The LAR proposes to revise TS Section 5.5.9.d.2 to add a one-time exception to the maximum inspection interval of 72 EFPM or every third refueling outage. The exception would require inspection at the fourth refueling outage (A1R23) after the last inspection (A1R19). TS Section 5.5.9.d.2 currently would require inspection at the third refueling outage (A1R22) in spring 2021. A1R23 will occur approximately 18 months later, in fall 2022 (approximately 70 EFPMs from the last inspection during A1R19). The proposed revised part of TS Section 5.5.9.d.2 reads as follows, with the bold portion being added for the one-time exception:

> For Unit 1, after the first refueling outage following SG installation, inspect each SG at least every 72 effective full power months or at least every third refueling outage (whichever results in more frequent inspections), **with the exception that each SG is to be inspected during the fourth refueling outage in A1R23 following inspections completed in refueling outage A1R19**. In addition, the minimum number of tubes inspected at each scheduled inspection shall be the number of tubes in all SGs divided by the number of SG inspection outages scheduled in each inspection period as defined in a, b, c, and d below. If a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at this location and that may satisfy the applicable tube plugging criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated. The fraction of locations to be inspected for this potential type of degradation at this location at the end of the inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the inspection period after the determination that a new form of degradation could potentially be occurring at this location divided by the total number of times the SG is scheduled to be inspected in the inspection period. Each inspection period defined below may be extended up to 3 effective full power months to include a SG inspection outage in an inspection period and the subsequent inspection period begins at the conclusion of the included SG inspection outage.

No changes to Unit 2 TS are requested. However, the amendment number for Unit 2 will be incremented because the Unit 1 and Unit 2 TS pages are common. This is an administrative change which does not affect the TS requirements. Based on the above, the NRC staff finds that incrementing the Unit 2 amendment number is acceptable.

3.3 NRC Staff Evaluation of Proposed TS Changes

3.3.1 Evaluation Approach

The licensee provided the following information to support its need for this one-time deferral of the upcoming SG inspection. In the LAR dated December 16, 2020, the licensee made the following statements:

> Circumstances are present in that the Centers for Disease Control and Prevention (CDC) and the State of Illinois have issued recommendations advising isolation activities (e.g., social distancing, group size limitations, self-quarantining, etc.) to prevent the spread of the COVID-19 virus. The nature of the SG inspections prevents compliance with the CDC and State of Illinois guidance for social distancing because workers must be in constant proximity to each other.
>
> As such, this request is being made to support Braidwood's proactive efforts to follow the CDC recommendations (e.g., social distancing, group size limitations, self-quarantining, etc.) by limiting the number of people onsite and in our neighboring communities.

The NRC staff's evaluation of the proposed one-time TS change was performed within the context of the COVID-19 pandemic and the potential impacts of this virus to plant personnel safety. Therefore, this staff evaluation should not be considered precedent setting for future routine plant amendments or generic industry licensing actions related to SG inspection intervals.

The NRC staff's evaluation of the proposed one-time TS change focused on the potential for affecting SG tube integrity, since maintaining SG tube integrity ensures the plant will meet its SG Program related TS, thereby protecting the public's health and safety. In particular, the staff evaluation assessed whether the LAR demonstrates that the structural integrity performance criterion (SIPC) and accident-induced leakage performance criterion (AILPC) will be met for cycle 23, which ends in fall 2022. These criteria are defined in TS Section 5.5.9.b.

The Braidwood Unit 1 inspections have detected tube degradation from fan bar, lattice grid, and FO wear, and the operational assessment (OA) provided by the licensee evaluates these as existing mechanisms using arithmetic deterministic analyses. These analyses use the worst-case single-tube analysis method of the Electric Power Research Institute (EPRI) Steam Generator Integrity Assessment Guidelines (IAG) (Reference 9) to provide a conservative estimate of the projected end-of-cycle (EOC) condition considering all uncertainties at a probability of 0.95 and at 50 percent confidence. The uncertainties used in the assessment are for the burst equation, the material strength, and the nondestructive examination flaw sizing technique. The single tube methods are referred to as "worst-case degraded tube" methods because the most severely flawed tube is selected for evaluation. The worst-case degraded tube OA methods involve selecting the most severely flawed tube at the beginning-of-cycle and applying conservative flaw growth over the intended inspection interval, to arrive at a predicted EOC flaw size and then determine if the SIPC and AILPC will be met at the EOC.

While tube-to-tube wear (TTW) has not occurred in the Braidwood Unit 1 SGs, TTW was evaluated as a potential mechanism using arithmetic deterministic analyses.

3.3.2 Evaluation of Existing Tube Degradation Mechanisms

Tube Wear at Fan Bars

The licensee's OA for fan bar wear was performed using the arithmetic approach described in the EPRI IAG. This approach adjusts the deepest fan bar wear flaw returned to service to account for eddy current sizing uncertainty, applies a 95th percentile growth rate, and then compares the projected flaw size at the next EOC inspection to the acceptable structural limits.

The A1R19 inspections for fan bar wear consisted of full-length bobbin probe examinations of 100 percent of the in-service tubes. There were 84 fan bar wear indications detected during A1R19 that ranged in depth from 3 to 16 percent through-wall (TW). The fan bar wear indications were sized using an EPRI qualified examination technique. No tubes were plugged during A1R19 due to fan bar wear. The largest fan bar wear indication allowed to remain in-service in A1R19 was 16 percent TW in SG 1D. The upper 95th percentile fan bar wear rate of 2.0 percent TW per effective full power year (EFPY) was applied and the wear depth at A1R23 (after four operating cycles) was projected to be 34.8 percent TW, which is less than the structural limit of 58.0 percent TW. The structural limit is calculated assuming a bounding wear length, applying a three times normal operating pressure differential (3xNOPD), and including material property and burst pressure equation uncertainties.

Wear at Lattice Grid Supports

The licensee's OA for lattice grid wear was performed using the arithmetic approach described in the EPRI IAG. This approach adjusts the deepest lattice grid wear flaw returned to service to account for eddy current sizing uncertainty, applies a 95th percentile growth rate, and then compares the projected flaw size at the next EOC inspection to the acceptable structural limits.

The A1R19 inspections for lattice grid wear consisted of full-length bobbin probe examinations of 100 percent of the in-service tubes. There were eleven lattice grid wear indications detected during A1R19 that ranged in depth from 2 to 9 percent TW. The lattice grid wear indications were sized using an EPRI qualified examination technique. No tubes were plugged due to lattice grid wear during A1R19. The largest lattice grid wear indications allowed to remain in-service in A1R19 were 9 percent TW in SG 1B. The upper 95th percentile lattice grid wear rate of 2.0 percent TW per EFPY was applied and the wear depth at A1R23 (after four operating cycles) was projected to be 28.0 percent TW, which is less than the structural limit of 56.6 percent TW (determined using 3xNOPD and including material property and burst pressure equation uncertainties).

Foreign Object Wear

In addition to wear at support structures, Braidwood Unit 1 has also experienced tube wear from FOs that have been transported into the SGs. The A1R19 inspections for FO wear consisted of full-length bobbin probe examinations of 100 percent of the in-service tubes, array probe examination of 955 periphery tubes, +Point probe examination of new FO wear indications with a two-tube bounding region, TTS inspection, and FOSAR. Thirteen indications of FO wear were identified during A1R19. The FO wear indications were sized using an EPRI qualified examination technique. Three tubes with four FO wear indications and one tube containing only a possible loose part signal (no wear) were preventatively plugged and stabilized because the FO (weld slag) could not be removed during FOSAR.

The tubes with the remaining nine FO wear indications identified during A1R19 were returned to service because the FOs were confirmed not to be present and the plugging criterion was not exceeded. The largest of these FO wear indications that was allowed to remain in-service was 25 percent TW in SG 1C. The structural limit for an axial flaw with limited circumferential extent and an assumed length of 2.0 inches is 57.6 percent TW (determined using 3xNOPD and including material property and burst pressure equation uncertainties). The licensee stated that the assumed length of 2.0 inches bounds the largest measured FO wear indication length. The nine FO wear indications identified during A1R19 in tubes that were returned to service will not grow to exceed the structural limit of 57.6 percent TW after four operating cycles because the FO that caused the wear is no longer present.

There are seven FOs remaining in the Braidwood Unit 1 SGs, three of the seven are legacy FOs. The licensee determined that these FOs are benign and significant wear due to these FOs is not projected through the next SG inspections. Table 4-1, "Priority 3 Foreign Objects Left in Service at A1R19," in Attachment 4 of the LAR provides additional information regarding the Priority 3 FOs remaining in the Braidwood Unit 1 SGs. The licensee projected the depth of the seven FOs remaining in the Braidwood Unit 1 SGs at A1R23 (after four operating cycles). The largest predicted depth for these FOs was 25.9 percent TW which is less than the structural limit of 57.6 percent TW.

The NRC staff finds the evaluation of existing FO wear acceptable since tubes with FOs remaining were either preventatively plugged or it was determined that the FOs would not cause significant wear through the next SG inspections, and tubes that no longer have a FO present will not incur additional wear and have been demonstrated to meet tube integrity.

The NRC staff also acknowledges that predicting future loose part generation is not possible since past fleet-wide operating experience has shown that new loose part generation, transport to the SG tube bundle, and interactions with the tubes cannot be reliably predicted. However, plants can reduce the probability of loose parts by maintaining robust foreign material exclusion programs and applying lessons learned from previous industry operating experience with loose parts. Plants in general, including Braidwood Unit 1, have demonstrated the ability to conservatively manage loose parts once they are detected by eddy current examinations or by secondary side FOSAR. If unanticipated aggressive tube wear from new loose parts should occur in a Braidwood Unit 1 SG, operating experience has shown that a primary-to-secondary leak is more likely to occur, rather than a loss of tube integrity. In the event of a primary-to-secondary leak, the NRC staff will interact with the licensee in accordance with established procedures in Inspection Manual Chapter (IMC) 0327, "Steam Generator Tube Primary-to-Secondary Leakage," dated January 1, 2019 (Reference 10), to confirm the licensee's conservative decision making.

Evaluation Summary of Existing Tube Degradation Mechanisms

The NRC staff finds the licensee's evaluation of tube wear at fan bar and lattice grid supports to be acceptable. Wear at these locations in the Braidwood Unit 1 SGs has been effectively managed since SG installation in 1998, without challenging tube integrity. The NRC staff finds the licensee's evaluation of FO wear to be acceptable since it accounts for tube wear from all known FOs within the SGs. During tube inspections, licensees perform condition monitoring to assess whether the measured flaw sizes are bounded by the previous OA flaw size projections. Table 5, "Summary of Condition Monitoring Performance for Existing Degradation Mechanisms during A1R19," in Attachment 1 of the LAR shows that the OA projected max depth from A1R17 (no SG tube inspections were performed during A1R17 but the OA was reviewed for any needed changes based on plant and industry operating experience since A1R16) was bounding for the measured tube wear mechanisms in A1R19, providing confidence that the OA methods and input assumptions can conservatively project future performance. The projections of the new OA submitted with the subject LAR continue to show that wear at support structures and FO wear will meet the structural limits for four cycles of operation and will not challenge the tube integrity limits. Therefore, the staff finds the evaluation of wear at support structures and FO wear to be acceptable.

Based on the evaluations, the NRC staff finds that the SIPC will be satisfied. Consistent with Reference 9, Section 4.2, "Tube Degradation Leakage Integrity," Attachment 4 in the LAR states that "…for volumetric degradation of the type and length detected or postulated to occur in the Braidwood Unit-1 SGs, the onset of pop-through leakage and burst are coincident. This means that if the degradation satisfies the structural integrity performance criteria (i.e., does not burst at 3xNOPD) then the degradation will not pop-through and leak at the much lower main steam line break pressure differential (i.e., MSLB PD) or the still lower normal operating pressure differential (i.e., NOPD)." Therefore, the NRC staff concludes that, by meeting the SIPC, the AILPC will also be satisfied.

Furthermore, the NRC staff finds that the licensee demonstrated that there is reasonable assurance that both the tube structural integrity and accident-induced leakage integrity performance criteria will be met for all tubes with existing known degradation until the A1R23.

3.3.3 Evaluation of Potential Tube Degradation Mechanisms

Tube-to-Tube Wear

The A1R19 inspections for TTW consisted of full-length bobbin probe examinations of 100 percent of the in-service tubes. Although no TTW has been identified in the Braidwood Unit 1 SGs or other recirculating SGs with similar design, the licensee assumed hypothetical TTW had occurred and was undetected during the last inspection. The largest TTW indication assumed to remain in-service in A1R19 was 20 percent TW, which is the depth at a probability of detection of 95 percent for the bobbin probe technique. The assumed wear rate for TTW was 2.0 percent TW per EFPY, which is the same as the wear rate for lattice grid and fan bar wear that has been identified in the Braidwood Unit 1 SGs. A flaw length of 30 inches was assumed for TTW and the corresponding structural limit was determined to be 55.5 percent TW (determined using 3xNOPD and including material property and burst pressure equation uncertainties). The wear depth at A1R23 (after four operating cycles) was projected to be 31.7 percent TW, which is less than the structural limit of 55.5 percent TW. Consistent with Reference 9, Section 4.2, "Tube Degradation Leakage Integrity," Attachment 4 in the LAR states that "…for volumetric degradation of the type and length detected or postulated to occur in the Braidwood Unit-1 SGs, the onset of pop-through leakage and burst are coincident. This means that if the degradation satisfies the structural integrity performance criteria (i.e., does not burst at 3xNOPD) then the degradation will not pop-through and leak at the much lower main steam line break pressure differential (i.e., MSLB PD) or the still lower normal operating pressure differential (i.e., NOPD)." Therefore, meeting the SIPC means that the AILPC will also be satisfied.

The NRC staff finds the licensee's evaluation of potential TTW to be acceptable because it conservatively assumed TTW was occurring and was undetected at the last inspection and was analyzed with a conservative arithmetic approach like the wear evaluations at structural supports and FO wear.

Based on the evaluation, the NRC staff finds that the licensee demonstrated that there is reasonable assurance that both the tube structural integrity and accident-induced leakage integrity performance criteria will be met for all tubes with potential degradation until the A1R23.

3.4 Mitigating Strategies

In the LAR, the licensee stated that as part of a mitigating strategy during operating cycle 23, it will implement a lower administrative limit on allowable primary-to-secondary leakage. The current TSs allow primary-to-secondary operational leakage of up to 150 gpd. The Exelon Generating Company, LLC, Primary-to-Secondary Leak Program procedure normally uses a more restrictive criterion of 100 gpd. For operating cycle 23, the licensee will lower the administrative limit to 30 gpd. The NRC staff has no objection to this administrative limit because the limit indicates a shutdown at a lower leakage level, which reduces the probability of a SG tube leak progressing to a SG tube burst. The NRC staff does not require the lower administrative limit as a condition of granting the amendment because, as discussed herein, the licensee demonstrated that tube integrity will be maintained until A1R23 and the NRC staff oversight activities, using IMC 0327, will confirm that that licensee responds appropriately should a reportable primary-to-secondary leak be detected.

3.5 Technical Evaluation Conclusion

Based on the information submitted, the NRC staff finds that the licensee has demonstrated there is reasonable assurance that the structural and leakage integrity of the Braidwood Unit 1 SG tubes will be maintained until the next SG tube inspections during refueling outage 23 in fall 2022. Therefore, the NRC staff approves the proposed changes to TS Section 5.5.9.d.2.

4.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION

The NRC's regulation in 10 CFR 50.92(c) states that the NRC may make a final determination, under the procedures in 10 CFR 50.91, that a license amendment involves no significant hazards consideration if operation of the facility, in accordance with the amendment, would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

An evaluation of the issue of no significant hazards consideration provided by the licensee is presented below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed one-time change will defer the SG inspection to be performed after four operating cycles. This change does not physically change the SGs, the plant, or the way the SGs or plant are operated. This change also does not change the design of the SG. Inspection frequencies and inspection activities are not an initiator to a SG tube rupture accident, or any other accident previously evaluated. As a result, the probability of an accident previously evaluated is not significantly increased. The SG tubes inspected by the SG Program continue to be required to meet the SG Program performance criteria and to be capable of performing any functions assumed in the accident analysis. As a result, the consequences of any accident previously evaluated are not significantly increased.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed one-time change will defer the SG inspection to be performed after four operating cycles. The proposed change does not alter the design function or operation of the SGs or the ability of an SG to perform its design function. The SG tubes continue to be required to meet the SG Program performance criteria. An analysis has been performed which evaluates all credible failure modes. This analysis resulted in no new or different kind of accident then has been previously evaluated. The proposed change does not create the possibility of a new or different kind of accident due to credible new failure mechanisms, malfunctions, or accident initiators that not considered in the design and licensing bases.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

   Response: No.

   The proposed one-time change will defer the SG inspection to be performed after four operating cycles. The proposed change does not change any of the controlling values of parameters used to avoid exceeding regulatory or licensing limits. The proposed change does not affect a design-basis or safety limit, or any controlling value for a parameter established in the UFSAR or the license.

   Therefore, the proposed change does not involve a significant reduction in a margin of safety.

The NRC staff reviewed the licensee's no significant hazards consideration analysis. Based on the review and on the NRC staff's evaluation of the underlying LAR as discussed above, the NRC staff concludes that the three standards of 10 CFR 50.92(c) are satisfied. In addition, the NRC staff determined that the incrementing of the Unit 2 amendment number is an administrative change that similarly meets the three standards in 10 CFR 50.92(c). Therefore, the NRC staff has made a final determination that no significant hazards consideration is involved for the proposed amendment and that the amendment should be issued as allowed by the criteria contained in 10 CFR 50.91.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Illinois State official was notified of the proposed issuance of the amendment on February 22, 2021. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to installation or use of a facility's components located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (86 FR 7116; January 26, 2021). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

7.0 CONCLUSION

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

8.0 REFERENCES

1. Murray, Dwi, Exelon Generation Company, LLC, letter to U.S. Nuclear Regulatory Commission, "Braidwood Station, Units 1 and 2, Application for Revision to TS 5.5.9, 'Steam Generator (SG) Program,' for a One-Time Deferral of Steam Generator Tube Inspections," dated December 16, 2020 (ADAMS Accession No. ML20351A433).

2. Murray, Dwi, Exelon Generation Company, LLC, letter to U.S. Nuclear Regulatory Commission, "Braidwood Station, Units 1 and 2, Response to Request for Additional Information Regarding Application for Revision to TS 5.5.9, 'Steam Generator (SG) Program,' for a One-Time Deferral of Steam Generator Tube Inspections," dated February 9, 2021 (ADAMS Accession No. ML21040A281).

3. U.S. Nuclear Regulatory Commission, Braidwood, Unit 1, Current Facility Operating License NPF-72, Tech Specs, Revised 10/09/2020 (ADAMS Accession No. ML053040362). (Note: Braidwood Units 1 and 2 share TS)

4. Exelon Generation Company, LLC, “Byron/Braidwood Stations, Revision 18 to Updated Final Safety Analysis Report, Chapter 3, Design of Structures, Components, Equipment, and Systems,” dated December 17, 2020 (ADAMS Accession No. ML19170A316).

5. Exelon Generation Company, LLC, “Byron/Braidwood Stations, Revision 18 to Updated Final Safety Analysis Report, Chapter 15, Accident Analysis,” dated December 17, 2020 (ADAMS Accession No. ML19170A347)

6. Enright, Daniel J., Exelon Generation Company, LLC, letter to U.S. Nuclear Regulatory Commission, “Braidwood Station, Unit 1 Steam Generator Tube Inspection Report for Refueling Outage 16,” dated August 17, 2012 (ADAMS Accession No. ML12249A057).

7. Marchionda-Palmer, Marri, Exelon Generation Company, LLC, letter to U.S. Nuclear Regulatory Commission, “Braidwood Station, Unit 1 – Steam Generator Tube Inspection Report for Refueling Outage 19,” dated February 27, 2017 (ADAMS Accession No. ML17058A085).

8. Wiebe, Joel S., U.S. Nuclear Regulatory Commission, letter to Michael J. Pacilio, Exelon Generation Company, LLC, “Braidwood Nuclear Power Station, Unit 1 – Review of the Spring 2012 Steam Generator Tube Inservice Inspections (TAC No. ME9505),” dated February 25, 2013 (ADAMS Accession No. ML13051B143).

9. Electric Power Research Institute, “Steam Generator Management Program: Steam Generator Integrity Assessment Guidelines,” Revision 4, dated June 2016 (ADAMS Accession No. ML16208A273; non-public, proprietary).

10. U.S. Nuclear Regulatory Commission, Inspection Manual Chapter (IMC) 0327, “Steam Generator Tube Primary-to-Secondary Leakage,” dated January 1, 2019 (ADAMS Accession No. ML18093B067).

Principal Contributor: L. Terry

Date of Issuance: March 10, 2021

SUBJECT: BRAIDWOOD STATION, UNITS 1 AND 2 - ISSUANCE OF AMENDMENT NOS. 220 AND 220 RE: A ONE-TIME DEFERRAL OF STEAM GENERATOR TUBE INSPECTIONS (EPID L-2020-LLA-0272) DATED MARCH 10, 2021

<u>DISTRIBUTION</u>:
PUBLIC
PM File Copy
RidsACRS_MailCTR Resource
RidsNrrDnrlNcsg Resource
RidsNrrDorlLpl3 Resource
RidsNrrDssStsb Resource
RidsNrrLASRohrer Resource
RidsNrrPMBraidwood Resource
RidsRgn3MailCenter Resource
LTerry, NRR

ADAMS Accession No.: ML21054A008

OFFICE	NRR/DORL/LPL3/PM	NRR/DORL/LPL3/LA	NRR/DNRL/NCSG/BC	NRR/DSS/STSB/BC
NAME	JWiebe	SRohrer(JBurkhardt for)	SBloom	VCusumano
DATE	2/23/2021	2/23/2021	2/19/2021	2/25/2021
OFFICE	OGC-NLO	NRR/DORL/LPL3/BC	NRR/DORL/LPL3/PM	
NAME	AGhosh-Haber	NSalgado	JWiebe	
DATE	3/5/2021	3/10/2021	3/10/2021	

OFFICIAL RECORD COPY