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

May 1, 2020

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, UNIT 2 - ISSUANCE OF AMENDMENT NO. 209 RE: ONE-TIME EXTENSION OF STEAM GENERATOR INSPECTIONS [COVID-19] (EPID L-2020-LLA-0069)

Dear Mr. Hanson:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 209 to Renewed Facility Operating License No. NPF-77 for the Braidwood Station, Unit 2. The amendment is in response to your application dated April 6, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20097J188), as supplemented by letters dated April 13, 2020 (ADAMS Accession No. ML20104C139) and April 16, 2020 (ADAMS Accession No. ML20107G585).

The amendment revises Technical Specification (TS) 5.5.9, "Steam Generator (SG) Program," for a one-time revision to the frequency for SG tube inspections. The requested amendment allows deferral of the required inspections until the next Braidwood Station, Unit 2, refueling outage. In addition, TS pages 5.5 – 8, 5.5 – 9, 5.5 – 10, and 5.5 – 11 were repaginated.

A copy of the Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's biweekly *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 No. STN 50-457

Enclosures:
1. Amendment No. 209 to NPF-77
2. Safety Evaluation

cc: Listserv

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. 209
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 April 6, 2020, as supplemented by letters dated April 13 and April 16, 2020, 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 renewed operating 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. 209 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 2 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Nancy L. Salgado

Digitally signed by Nancy L. Salgado
Date: 2020.05.01 13:01:06 -04'00'

Nancy L. Salgado, 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: May 1, 2020

ATTACHMENT TO LICENSE AMENDMENT NO. 209

RENEWED FACILITY OPERATING LICENSE NO. NPF-77

BRAIDWOOD STATION, UNIT 2

DOCKET NO. STN 50-457

Replace the following pages of the Renewed Facility Operating License 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 No. NPF-77	License No. NPF-77
Page 3	Page 3
TSs	TSs
5.5 – 8	5.5 – 8
5.5 – 9	5.5 – 9
5.5 – 10	5.5 – 10
5.5 – 11	5.5 – 11

(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. 209 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 Programs and Manuals

5.5.9 Steam Generator (SG) Program (continued)

2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed a total of 1 gpm for all SGs.

3. The operational LEAKAGE performance criteria is specified in LCO 3.4.13, "RCS Operational LEAKAGE."

c. Provisions for SG tube plugging criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal wall thickness shall be plugged. The following alternate tube plugging criteria shall be applied as an alternative to the 40% depth based criteria:

For Unit 2, tubes with service-induced flaws located greater than 14.01 inches below the top of the tubesheet do not require plugging. Tubes with service-induced flaws located in the portion of the tube from the top of the tubesheet to 14.01 inches below the top of the tubesheet shall be plugged upon detection.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube plugging criteria. For Unit 2, portions of the tube below 14.01 inches from the top of the tubesheet are excluded from this requirement.

5.5 Programs and Manuals

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). 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.

5.5.9 Steam Generator (SG) Program (continued)

a) After the first refueling outage following SG installation, inspect 100% of the tubes during the next 144 effective full power months. This constitutes the first inspection period;

b) During the next 120 effective full power months, inspect 100% of the tubes. This constitutes the second inspection period;

c) During the next 96 effective full power months, inspect 100% of the tubes. This constitutes the third inspection period; and

d) During the remaining life of the SGs, inspect 100% of the tubes every 72 effective full power months. This constitutes the fourth and subsequent inspection periods.

3. For Unit 2, after the first refueling outage following SG installation, inspect each SG at least every 48 effective full power months or at least every other refueling outage (whichever results in more frequent inspections), with the exception that each SG is to be inspected during the third refueling outage in A2R22 following inspections completed in refueling outage A2R19. 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, and c 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

5.5.9 Steam Generator (SG) Program (continued)

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.

a) After the first refueling outage following SG installation, inspect 100% of the tubes during the next 120 effective full power months. This constitutes the first inspection period;

b) During the next 96 effective full power months, inspect 100% of the tubes. This constitutes the second inspection period; and

c) During the remaining life of the SGs, inspect 100% of the tubes every 72 effective full power months. This constitutes the third and subsequent inspection periods.

4. For Unit 1, if crack indications are found in any SG tube, then the next inspection for each affected and potentially affected SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever results in more frequent inspections). For Unit 2, if crack indications are found in any SG tube from 14.01 inches below the top of the tubesheet on the hot leg side to 14.01 inches below the top of the tubesheet on the cold leg side, then the next inspection for each affected and potentially affected SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever results in more frequent inspections).

   If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 209 TO RENEWED

FACILITY OPERATING LICENSE NO. NPF-77

EXELON GENERATION COMPANY, LLC

BRAIDWOOD STATION, UNIT 2

DOCKET NO. STN 50-457

1.0 INTRODUCTION

By letter April 6, 2020 (Agencywide Documents and Access Management System (ADAMS) Accession No. ML20097J188), Exelon Generation Company, LLC (the licensee) requested a one-time change to the technical specifications (TSs) for Braidwood Station, Unit 2 (Braidwood Unit 2). The licensee provided additional information supporting its request in letters dated April 13, 2020 (ADAMS Accession No. ML20104C139), and April 16, 2020 (ADAMS Accession No. ML20107G585). Attachments 2 and 5 of the April 16, 2020, letter contain sensitive unclassified non-safeguards information and, accordingly, are withheld from public disclosure pursuant to Section 2.390, “Public inspections, exemptions, requests for withholding,” of Title 10 of the *Code of Federal Regulations* (10 CFR).

The proposed one-time change would revise the frequency of the required steam generator tube inspections contained in TS 5.5.9.d.3, “Steam Generator (SG) Program,” to allow deferral of inspections until the next outage. 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 the changes to be approved as an emergency license amendment in accordance with 10 CFR 50.91(a)(5). Based on an initial review of the April 6, 2020, letter, the U.S. Nuclear Regulatory Commission (NRC) staff determined that the license amendment request (LAR) could be processed under 10 CFR 50.91(a)(6) as an exigent amendment because there was time for a 14-day comment period prior to issuing the amendment.

The supplemental letters dated April 13 and April 16, 2020, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff’s original proposed no significant hazards consideration determination as published in the *Federal Register* on April 16, 2020 (85 FR 21274).

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 serve to isolate radiological fission products in the primary coolant from the secondary coolant and the environment. As used in this safety evaluation, SG tube integrity means that no materials or other degradation would hinder the capability of tubes to perform this safety function in accordance with the plant design and licensing basis.

2.2 Regulatory Requirements and Guidance

Requirements for the integrity of the SG tubing are established by NRC regulations. Specifically, general design criterion (GDC) 14, “Reactor coolant pressure boundary,” in Appendix A to 10 CFR Part 50, “General Design Criteria for Nuclear Power Plants,” requires 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.” GDCs 15, “Reactor coolant system design,” and 31, “Fracture prevention of reactor coolant pressure boundary,” require an RCPB design with sufficient margin to ensure design pressure is not exceeded and to ensure that RCPB materials behaves in a non-brittle manner. GDC 31 also requires that the probability of a rapidly propagating fracture be minimized. The TSs for all pressurized-water reactor plants require an SG program to be established and implemented to ensure that SG tube integrity is maintained. The SG program and other programs that the licensee must establish and maintain to assure that the facility is operated in a safe manner are listed in the administrative controls section of the TSs. For Braidwood, Unit 2, the requirements for performing SG tube inspections and repair are in TS 5.5.9, while the requirements for reporting the SG tube inspections and repair are in TS 5.6.9.

For Braidwood, Unit 2, SG tube integrity is maintained by meeting the performance criteria specified in TS 5.5.9.b for structural and leakage integrity, consistent with the Braidwood Updated Final Safety Analysis Report (UFSAR), Section 5.4.2 (ADAMS Accession No. ML18355A425). TS 5.5.9.a requires that a condition monitoring (CM) assessment be performed during each outage in which the SG tubes are inspected, to confirm that the performance criteria are being met. TS 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 repair criteria. The applicable tube repair criteria, specified in TS 5.5.9.c, are that tubes found during inservice inspection to contain flaws with a depth equal to or exceeding 40 percent of the nominal wall thickness shall be plugged, unless the tubes are permitted to remain in service through application of alternate repair criteria provided in TS 5.5.9.c.

Braidwood, Unit 2, TS 3.4.13 includes a limit on operational primary-to-secondary leakage, beyond which the plant must be shut down. Should an existing flaw that exceeds the tube integrity repair limit not be detected during the periodic tube surveillance required by the plant TS, the operational leakage limit provides added assurance of timely plant shutdown before tube structural and leakage integrity are impaired.

In Braidwood UFSAR Chapter 15 (ADAMS Accession No. ML19170A347), the licensee analyzed the consequences of postulated design-basis accidents, such as a SG tube rupture and a steam line break. These analyses consider primary-to-secondary leakage that may occur during these events and must show that the offsite 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 2, are being changed by the proposed amendment and, 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.

The regulation, 10 CFR 50.36, “Technical specifications,” requires that each applicant for an operating license include (in its application) proposed TSs and establishes the regulatory requirements for the content of TSs. The TS are required to be derived from the analyses and evaluation included in the safety analysis report, and amendments thereto, submitted pursuant to 10 CFR 50.34, “Contents of applications; technical information.” The Commission may include such additional technical specifications as the Commission finds appropriate.

Section 10 CFR 50.36(c) specifies the categories of items that are to be included in the TSs, including (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; and (5) administrative controls. This LAR, as supplemented, proposed changes to TS 5.5.9, which provides plant-specific details on the Braidwood, Unit 2, SG inspection program. The NRC staff reviewed the LAR to determine if the licensee’s modified SG inspection program complies with the 10 CFR 50.36(c)(5), which states, “Administrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner.”

3.0 TECHNICAL EVALUATION

3.1 Background

3.1.1 Steam Generator Design

Braidwood, Unit 2, has four Westinghouse Model D5 SGs, which have been in service since the plant began commercial operation in 1988. Each SG contains 4,570 thermally-treated Alloy 600 tubes (Alloy 600TT) with a nominal outside diameter of 0.750 inches and a nominal wall thickness of 0.043 inches. The tubes are hydraulically expanded at each end for the full depth of the tubesheet and are welded to the primary face of the tubesheet. The straight length of the tubes is supported by Type 405 stainless steel tube support plates (TSPs) with quatrefoil broached-holes. The U-bend regions of the tubes are supported by two sets of chrome-plated Alloy 600 V-shaped anti-vibration bars (AVBs). To reduce residual stress, the U-bend region of the tubes in Rows 1-9 (short radius) were stress-relieved after bending.

The SGs have a preheater region where most of the incoming feedwater is directed through a series of baffle plates before entering the main region of the tube bundle. The preheater baffle plates are Type 405 stainless steel with drilled holes. Approximately 124 tubes are expanded into the drilled holes in the second and third cold-leg baffle plates to limit tube vibration.

3.1.2 Operating Experience

All four Braidwood, Unit 2, SGs were inspected in the Braidwood, Unit 2, Refueling Outage 17, (A2R17) in the spring of 2014 and in A2R19 in the spring of 2017. The LAR discussed information regarding the SG inspections at Braidwood, Unit 2, including the eddy current examinations performed, documented in the spring 2014 and spring 2017 SG tube inspection reports dated November 14, 2014 (ADAMS Accession No. ML14318A211), and August 24, 2017 (ADAMS Accession No. ML17236A457).

Since the SGs were placed in service in 1988, a total of 289 tubes have been plugged (107 in SG A, 66 in SG B, 72 in SG C, and 44 in SG D). Table 2 of Attachment 1 to the licensee's letter dated April 6, 2020, identifies the causes for tube plugging and the corresponding number of tubes plugged.

SGs with Alloy 600TT tubes are known to have some tubes with higher residual stress that are more susceptible to stress corrosion cracking (SCC). At Braidwood, Unit 2, axial outside diameter SCC (ODSCC) has been detected in five high stress tubes. Additional detail about the SCC experience is discussed below and in Section 3.3.2 of this safety evaluation.

Braidwood, Unit 2, has the following existing degradation mechanisms: axial ODSCC at hot-leg TSPs and in a freespan ding (one indication), AVB wear, wear at TSP quatrefoil openings, wear at drilled holes in preheater baffle plates (also called drilled support plates (DSPs)), and wear from foreign objects (FOs). Inspections during A2R19 reported a total of 1,128 wear indications from all mechanisms in the four Braidwood Unit 2 SGs. No forms of corrosion degradation, including SCC, were detected during the A2R19 inspections. Table 1 summarizes the historical indications of ODSCC, which are all axial in orientation and located at TSPs (except for one at a freespan ding as shown). Table 2 summarizes the wear indications by location reported from A2R19, the latest inspection.

Table 1: Axial ODSCC Indications in Braidwood, Unit 2, SG Tubes

Year	**Outage**	**SG**	**Tube (Row-Column)**	**Affected TSP**
2003	A2R10	A	R25-C42	03H
2003	A2R10	C	R21-C50	03H
2003	A2R10	C	R21-C50	05H
2003	A2R10	C	R38-C20	07H
2011	A2R15	D	R2-C35	03H
2011	A2R15	D	R2-C35	07H
2011	A2R15	D	R2-C35	09H
2012	A2R16	C	R44-C47	03H
2012	A2R16	C	R44-C47	05H
2012	A2R16	C	R44-C47	03H/05H*

*This indication was at a ding in the freespan between 03H and 05H

Table 2: Wear Indications Reported in Braidwood, Unit 2, SGs (A2R19, Spring 2017)

Wear Location	**Total Number of Indications in Each SG**			
	SG A	**SG B**	**SG C**	**SG D**
AVB	464	126	317	198
TSP Quatrefoil	1	0	0	1
TSP Drilled hole	0	3	0	0
Secondary side foreign object	7	7	0	4

Secondary side inspections were performed in all four SGs in the spring of 2014 and in the spring of 2017. Inspections in 2014 (A2R17) included visual and ultrasonic inspections of the moisture separator regions in SG A and SG B, preheater/waterbox inspections of specific high-flow regions in SG C and SG D, upper bundle visual inspections in SG C, and foreign object search and retrieval in SG D. Inspections in 2017 (A2R19) included visual and ultrasonic inspection of the steam drum SG C and SG D, preheater/waterbox inspections of specific high-flow regions in SG A and SG C, upper bundle visual inspection in SG C, and sludge lancing and foreign object search and retrieval in all four SGs. Seven new indications of foreign object wear on seven tubes were identified in 2017. Four tubes were plugged in 2017 due to foreign object wear (one tube) or possible loose parts (three tubes).

Only leakage greater than 3 gallons per day (gpd) is required by TS 5.6.9.h to be included in the Steam Generator Tube Inspection Report. The licensee stated in its letter dated April 6, 2020, that all primary-to-secondary leakage trends noted during the past two operating cycles (20 and 21) are below 3 gpd.

3.2 Proposed TS Changes

3.2.1 Current TS Requirements

Braidwood, Unit 2, TS 5.5.9.d.3 states, in part:

> For Unit 2, after the first refueling outage following SG installation, inspect each SG at least every 48 effective full power months or at least every other 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, and c below.

Braidwood, Unit 2, TSs 5.5.9.d.3.a and 5.5.9.d.3.b define the SG tube inspection requirements for the first and second inspection periods after SG installation. TS 5.5.9.d.3.c defines the SG tube inspection requirements for the third and subsequent inspection periods (remaining life of the SGs). Specifically, TS 5.5.9.d.3.c states:

> During the remaining life of the SGs, inspect 100% of the tubes every 72 effective full power months. This constitutes the third and subsequent inspection periods.

Braidwood, Unit 2, is currently in the fourth inspection period; therefore, the TS requires 100 percent of the tubes to be inspected every 72 effective full power months.

In addition to the above TS change, the licensee proposes to repaginate TS pages 5.5 – 8, 5.5 – 9, 5.5 – 10, and 5.5 – 11.

3.2.2 Description of Proposed TS 5.5.9.d.3 Changes

The LAR proposes to revise TS 5.5.9.d.3 to add a one-time exception to the maximum inspection interval of 48 months or every other refueling outage. The exception would require inspection at the third refueling outage (A2R22) after the last inspection (A2R19). TS 5.5.9.d.3 currently requires an inspection during Refueling Outage 21 (A2R21), which began in April 2020. Refueling Outage 22 (A2R22) will occur approximately 18 months later, in the fall of 2021. The proposed revised part of TS 5.5.9.d.3, with the underlined portion being added for the one-time exception, reads as follows:

> For Unit 2, after the first refueling outage following SG installation, inspect each SG at least every 48 effective full power months or at least every other refueling outage (whichever results in more frequent inspections), with the exception that each SG is to be inspected during the third refueling outage in A2R22 following inspections completed in refueling outage A2R19.

3.3 Staff Evaluation of Proposed TS Changes

3.3.1 Evaluation Approach

The NRC staff evaluated the LAR to determine if the modified TS Administrative controls are consistent with the regulations, guidance, and licensing and design basis.

The licensee did not request any change to its current TS limiting conditions for operation and surveillance requirements, and states in its letter dated April 6, 2020, "All applicable regulatory requirements will continue to be satisfied as a result of the proposed license amendment."

The NRC staff evaluation of the proposed one-time TS changes focused on the potential to affect SG tube integrity because maintaining SG tube integrity ensures the plant will meet its SG program-related TS, thereby protecting the public's health and safety. Specifically, the staff evaluation assessed whether the LAR demonstrates that the structural integrity performance criterion and accident-induced leakage performance criterion will be met for Cycle 22, which terminates at the fall 2021 refueling outage (A2R22). These criteria are defined in TS 5.5.9.b.

The Braidwood, Unit 2, SGs have experienced tube degradation from wear against support structures and from axial ODSCC in tubes with potentially high residual stress; these degradation mechanisms are referred to as existing mechanisms. The SCC mechanisms that have not been experienced in the Braidwood, Unit 2, SGs are referred to as potential mechanisms. In its April 6, 2020, letter, the licensee stated that existing wear mechanisms were analyzed in a deterministic manner while both existing and potential SCC mechanisms were analyzed in a full-bundle probabilistic manner. The analyses of the existing and potential degradation mechanisms are discussed below.

3.3.2 Evaluation of Existing Tube Degradation Mechanisms

Tube Wear at Anti-Vibration Bars

The A2R19 inspections for AVB wear consisted of full-length bobbin probe examinations of 100 percent of the in-service tubes. The largest AVB indications allowed to remain in-service in A2R19 were 39 percent through-wall (TW). The upper 95th percentile AVB wear rate of 2.9 percent TW per effective full power year (EFPY) was applied and the projected wear depth at A2R22 (after three operating cycles) was 52 percent TW, which is less than the CM limit of 69 percent TW.

Wear at Tube Support Plates and Drilled Support Plates

The A2R19 inspections for TSP and DSP wear consisted of full-length bobbin probe examinations of 100 percent of the in-service tubes. In the Braidwood, Unit 2, SGs, there are only two wear indications at TSPs and only three wear indications at DSPs. The largest indications returned to service in A2R19 were 28 percent TW and 5 percent TW, for TSPs and DSPs, respectively. Due to the small population of TSP and DSP indications in the SGs, there was not sufficient data to develop accurate growth rate curves, so analyses of these mechanisms applied the maximum observed growth rate (instead of the 95th percentile growth rate) to the largest flaw left in service during the last inspection. For TSPs, a growth rate of 2.1 percent TW/EFPY was applied and the projected wear depth at A2R22 (after three operating cycles) was 37 percent TW, which is less than the CM condition monitoring limit of 49 percent TW. For DSPs, a growth rate of 2.1 percent TW/EFPY was applied and the projected wear depth at A2R22 (after three operating cycles) was 14 percent TW, which is less than the CM limit of 55 percent TW.

Evaluation Summary for Wear at AVBs, TSPs, and DSPs

Wear at these locations has been effectively managed for many cycles without challenging tube integrity. During SG tube inspections, licensees perform CM to assess whether the measured flaw sizes are bounded by the previous operational assessment (OA) flaw size projections. During the most recent Braidwood, Unit 2, inspection in A2R19, the OA worst case projections from A2R17 were bounding for the measured tube wear mechanisms in A2R19, 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 AVB, TSP, and DSP structural wear mechanisms will meet the CM limits for three cycles of operation and will not challenge the tube integrity limits. Therefore, the NRC staff finds the licensee's evaluation of tube wear at AVBs, TSPs, DSPs to be acceptable.

Foreign Object (FO) Wear

In addition to wear at support structures, Braidwood, Unit 2, has also experienced tube wear from FOs that have been transported into the SGs. After A2R19, a separate two-cycle OA specifically covering potential future FO wear was developed. To address FO wear for an additional cycle of operation, the licensee revised the separate two-cycle OA to address a third cycle of operation (see Attachment 6 of the licensee's letter dated April 16, 2020). The assessment addressed the foreign objects identified during the A2R19 FO search and retrieval inspections, as well as the newly reported volumetric wear indications in the upper tube bundle. The newly identified volumetric indications were examined by +Point™ probe and confirmed that no FOs remained at the location of the volumetric indication. Three adjacent tubes with a possible loose part indication were in a location that could not be accessed on the secondary side, so the three tubes were preventively plugged.

The licensee determined that the potential for mechanical wear or impact damage from the FOs remaining inside the SGs for three operating cycles was low, based on the size of the non-fixed FOs being smaller and having less mass than known bounding objects. The known objects were determined by operating experience, thermal-hydraulic analysis, and modelling, not to cause wear or impact damage. In addition, the small FOs identified during A2R19 were determined to have insufficient mass to affect other systems, structures or components that are connected to the steam generators. Thus, the FOs would not result in mechanical interferences with any active components and would not adversely affect systems upstream or downstream of the SGs during the proposed three operating cycles.

The NRC staff finds the licensee's evaluation of FOs to be acceptable since it accounts for tube wear from known FOs within the SGs. Based on this analysis, the licensee demonstrated that tube integrity will be maintained until A2R22.

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. Plants in general, including Braidwood, Unit 2, have demonstrated the ability to conservatively manage loose parts once they are detected by eddy current examinations or by secondary-side FO search and retrieval inspections. If unanticipated aggressive tube wear from new loose parts should occur in a Braidwood, Unit 2, 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 ensure that the licensee is responding to primary-to-secondary leakage in a conservative manner that is consistent with the industry initiative NEI 97-06, "Steam Generator Program Guidelines," Revision 3 (ADAMS Accession No. ML111310708) as specified by Inspection Manual Chapter (IMC) 0327, "Steam Generator Tube Primary-to-Secondary Leakage," dated January 1, 2019 (ADAMS Accession No. ML18093B067).

In its letter dated April 6, 2020, the licensee states that as part of a mitigating strategy during Operating Cycle 22, it will implement a lower administrative limit on allowable primary-to-secondary leakage. The current TS allow primary-to-secondary operational leakage of up to 150 gpd). In its letter dated April 6, 2020, the licensee states that for Operating Cycle 22, its administrative limit will be lowered from 100 gpd 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 here-in, the licensee demonstrated that tube integrity will be maintained until A2R22 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.

<u>Axial ODSCC in High Residual Stress Tubes</u>

The existence of Alloy 600TT tubes with potentially high residual stresses was first noted in industry in 2002 and is discussed in more detail in NRC Information Notice 2002-21, "Axial Outside-Diameter Cracking Affecting Thermally Treated Alloy 600 Steam Generator Tubing," dated June 25, 2002 (ADAMS Accession No. ML021770094).

As noted previously in Table 1 of this safety evaluation, Braidwood, Unit 2, has experienced axial ODSCC in high stress tubes in three previous inspections; a brief summary is provided here:

Prior to A2R10 in 2003, the Braidwood, Unit 2, licensee reviewed previous eddy current inspection data and identified 77 tubes in the Braidwood, Unit 2, SGs that potentially had higher residual stress from manufacturing. During the A2R10 inspections, three high-row tubes were identified with axial ODSCC indications at TSPs. The three tubes were plugged and, in addition, three low-row high stress tubes without indications were plugged. During the A2R15 inspections in 2011, three axial ODSCC indications were found in a tube with potentially high residual stress, so the tube was plugged and removed from service. During the A2R16 inspections in 2012, three axial ODSCC indications were identified in a tube not previously identified as potentially having high residual stress. Upon review of the tube's eddy current signature, it was determined that this high-row tube had been missed in the initial screening process. The tube was plugged and removed from service. To address the missed tube in the initial screening process, all high-row tubes were re-evaluated in A2R16. No additional tubes with potentially high residual stress were identified. Also, in A2R16, the remaining tubes with potentially high residual stress were reviewed for eddy current signatures like the tube with axial ODSCC that was found in A2R16. As a result of this review, two tubes were identified with eddy current signatures like the tube with axial ODSCC that was identified in A2R16. These two tubes did not have any indications of axial ODSCC, but were preventively plugged. Additional details are contained in NUREG-2188, "U.S. Operating Experience with Thermally Treated Alloy 600 Steam Generator Tubes Through December 2013," dated February 2016 (ADAMS Accession No. ML16061A159).

In A2R19, the licensee performed the following inspections to detect axial ODSCC in the 68 in-service tubes identified as high residual stress tubes:

- 100 percent full length bobbin inspections.
- To improve the probability of detection for axial ODSCC, starting in A2R17 (March 2014) and continued in A2R19 (April 2017), all TSPs with a mixed residual of 0.4 maximum vertical volts (V_{vm}) were tested with a +Point™ probe.
- 100 percent of all hot-leg and cold-leg TSP intersections with an X-Probe™.
- All dents and dings greater than 2 volts were tested with a +Point™ probe.

No axial ODSCC indications were reported by the licensee in either the A2R17 or A2R19 SG inservice inspections.

To address the possibility that other non-detected high stress tubes are still present, the licensee used two OA models in the evaluation of the SGs. The first model is an acute initiation model that closely mimics the prior history of the Braidwood, Unit 2, SGs by introducing a discrete number of flaws into the model in a brief period. The second model is a low Weibull slope model that introduces flaws on a more traditional basis (i.e., spread out over time). The population size used in the low Weibull slope model assumes there are significantly more high stress tubes than the 68 known high stress tubes, to address the potential for additional high stress tubes existing in the SGs. The upper bound default growth rate from the Electric Power Research Institute, Steam Generator Management Program: "Steam Generator Integrity

Assessment Guidelines," Revision 4, dated June 2016 [1] (EPRI IAGL), is applied for both high stress tube models. The analysis of axial ODSCC at freespan dings on high residual stress tubes is also addressed by the low slope Weibull analysis model. Therefore, the analyses of axial ODSCC on high residual stress tubes did not require combination of probabilities.

Full-bundle probabilistic analyses were used to calculate the probability of tube burst and accident-induced leakage potential, in accordance with Section 8.3 of the EPRI IAGL. The probabilistic models included the important input distributions for; material strength properties of the tubing, probability of detection for the eddy current inspection technique, a lognormal crack growth rate model appropriate for each mechanism at T_{Hot}, and the use of a Weibull initiation function predicting when SCC flaws have developed over time. One important feature built into the model is its ability to predict and account for the cumulative effect of a population of newly initiated SCC indications and preexisting undetected SCC indications that were either missed or too small to be detected by the eddy current technique used. The probability of burst met the acceptance criteria, as specified in the licensee's letter dated April 16, 2020, of less than 5% for both models and therefore provides reasonable assurance that the structural integrity performance criteria of TS 5.5.9.b.1 as well as the accident-induced leakage performance criteria of TS 5.5.9.b.2 are met.

The NRC staff reviewed the licensee's probabilistic evaluations of existing axial ODSCC mechanisms, including the licensee's April 13, 2020, letter response to the NRC staff's request for additional information. The number of tubes with axial ODSCC detected in the Braidwood, Unit 2, SGs is small and the licensee has taken increased inspection measures over the last two refueling inspections to better detect any additional indications. The NRC staff considers the increased inspection measures taken by the licensee and the number of tubes assumed with axial ODSCC in models used for the probabilistic evaluations to be conservative for Braidwood, Unit 2, because the actual number of tubes with axial ODSCC is small, the licensee has taken increased inspection measures over the last two refueling inspections to better detect any additional indications, and the licensee has assumed for the purposes of the model that previous inspections have missed some tube indications and some tubes will develop indications prior to the A2R22 outage. The licensee implemented two probabilistic evaluations for the axial ODSCC and both models indicate that the calculated probability of burst and accident-induced leakage for axial ODSCC satisfy the structural integrity performance criterion and the accident-induced leakage performance criterion of TS 5.5.9.b.1 and TS 5.5.9.b.2, respectively, until A2R22 in the fall of 2021. Based on the above, the NRC staff finds that there is reasonable assurance that the performance criterion of TS 5.5.9.b.1 and TS 5.5.9.b.2, will be met, until A2R22 in the fall of 2021.

3.3.3 Evaluation of Potential Tube Degradation Mechanisms

In addition to existing tube degradation mechanisms, the licensee considered potential degradation mechanisms. By letter dated October 5, 2012 (ADAMS Accession No. ML12262A360), the NRC staff approved an H*[2] amendment for Braidwood, Unit 2. The letter concluded that potential tube degradation below the H* depth in the tubesheet does not affect tube integrity. Therefore, this safety evaluation does not consider potential tube degradation between the H* distance and the tube end.

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[1] Available from Electric Power Research Institute, 3420 Hillview Avenue, Palo Alto, California 94304.

[2] H* is the minimum engagement distance between the tube and tubesheet, measured downward from the top of the tubesheet, that is proposed, as needed, to ensure the structural and leakage integrity of the tube to tubesheet joints.

Some plants with SGs with Alloy 600TT tubes have experienced SCC initiating from either the primary side of the tube (primary-water stress corrosion cracking, PWSCC) or outside diameter of the tube (ODSCC). This has occurred at multiple locations on the tubes. The SCC mechanism is known to be dependent on temperature. In general, plants operating at higher temperatures are more prone to SCC compared to plants operating at lower temperature. Similarly, in general, hotter portions of the tubes are more susceptible to SCC than colder sections of the tubes. Higher residual stress is also an accelerating factor for SCC in SG tubes. SGs with Alloy 600TT tubes are known to have some tubes with higher residual stress that are more susceptible to cracking. Braidwood, Unit 2, has identified 68 tubes with potentially higher residual stress, as previously discussed in Section 3.3.2 of this safety evaluation.

Although there are many forms of SCC that have not been detected in the Braidwood, Unit 2, SG tubes, eddy current examinations with specialized probes are performed to detect cracking if it forms. The licensee performed the following eddy current inspections in A2R19 and A2R17 to detect potential forms of SCC not previously identified at Braidwood, Unit 2:

- Axial ODSCC at tube dents and dings – Bobbin inspections of 100 percent of in-service tubes for detection at dents and dings up to 5 volts. Also, a 50 percent sample of dents and dings greater than 5 volts in the hot-leg, U-bend, and outside the preheater were inspected with a +Point™ probe. In the preheater and flow distribution baffle on the hot-leg, a 50 percent sample of dents and dings greater than 2 volts were inspected with a +Point™ probe. The remaining 50 percent sample was inspected in A2R17 (March 2014).

- Circumferential and Axial ODSCC and PWSCC at the hot-leg top of tubesheet (TTS) expansion transition (including overexpansions) and inside the tubesheet (including bulges) to the H* depth – A 63 percent Array probe inspection was performed (50 percent inner bundle plus 3 tubes deep on the periphery). The remaining 37 percent inspection was performed in A2R17.

- Axial PWSCC in small radius U-bends – A 50 percent sample of the Row 1 and Row 2 U-bends with a +Point™ probe was performed. The remaining 50 percent was inspected in A2R17. In addition, all tubes with the "Blairsville Bump" manufacturing anomaly[3] were inspected with a +Point™ probe in A2R17 and A2R19.

The OA of potential degradation mechanisms projects the behavior of postulated flaws that could have been present at or prior to the last SG inspection in A2R19 and those that could initiate during the three-cycle operating period ending at A2R22. The OA of potential SCC mechanisms made the following conservative assumptions:

- All potential SCC mechanisms are assumed to be existing and are evaluated in the OA.

- Prior to the most recent tube inspection, SCC had initiated and was missed during the most recent inspection. This assumption creates a population of undetected flaws that exist at the start of the cycle prior to the most recent inspection.

- Typical default crack growth rates from the EPRI IAGL for circumferential ODSCC at hot-leg TTS expansion transitions and axial ODSCC at dents and dings, were applied.

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[3] More information on this manufacturing anomaly is available in NUREG-2188.

- For mechanisms that were sampled in A2R19, the tube population was divided into two groups per the implemented sampling plan (inspected and non-inspected) in accordance with Section 8.6 of the EPRI IAGL. The probability of burst and leakage assessment was individually computed for each partially inspected group and later numerically combined to give the total tube bundle probabilities for the mechanism.

- All OA analyses were performed using a bounding SG approach where assumed susceptible populations are represented by the largest number of locations for any SG.

- Axial ODSCC at TSP intersections on non-high residual stress tubes was reported for the first time in Alloy 600TT tubes during the fall of 2019. This mechanism was addressed with a low Weibull slope initiation model and the assumption that the non-high stress tubes would be expected to experience growth rates bounded by the EPRI IAGL typical default value.

These potential mechanisms were each evaluated by performing full-bundle probabilistic analyses to calculate the probability of tube burst and accident-induced leakage potential, in accordance with Section 8.3 of the EPRI IAGL. The probabilistic models included the important input distributions for material strength properties of the tubes, probability of detection for the eddy current inspection technique, a lognormal crack growth rate model appropriate for each mechanism at T_{Hot}, and the use of a Weibull initiation function predicting when SCC flaws have developed over time. One important feature built into the model is its ability to predict and account for the cumulative effect of a population of newly initiated SCC indications and preexisting undetected SCC indications that were either missed or were too small to be detected by the eddy current technique used.

The NRC staff reviewed the licensee's probabilistic evaluation of potential mechanisms. The staff considers the probabilistic evaluation assumptions to be conservative for Braidwood, Unit 2 based on the following. Although only axial ODSCC in tubes with potentially high residual stress has been detected in inservice inspections, the model assumes that other forms of SCC existed at the previous inspection, were missed, and grow until the A2R22 inservice inspections. The model predicts that axial and circumferential cracks will be detected at the TTS, axial cracks will be detected at the TSP intersections, and axial cracks will be detected at dents/dings at the A2R22 inspection. The calculated probability of burst for all potential mechanisms satisfy the structural integrity performance criterion margin requirements until the A2R22 inspections. The accident-induced leakage was determined by summing the projected leak rates at the start of A2R22. Based on past operating experience at Braidwood, Unit 2, it would not be credible to assume that all potential mechanisms would be active in one operating period. Assuming three limiting mechanisms become active in one SG (i.e., axial ODSCC at TSPs, circumferential ODSCC at TTS, and axial ODSCC at dings/dents), the cumulative leak rate is determined to be 0.11 gallons per minute. This leakage value is less than the accident-induced leakage performance criterion leak limit of 0.5 gallons per minute for any one SG as stated in the licensee's letter dated April 16, 2020, based on the guidance in NEI 97-06 and the TS 5.5.9.b.2 requirements. The accident-induced leakage performance criterion is also satisfied when assuming the three limiting SCC mechanisms are active during Operating Cycle 22.

Therefore, based on the evaluation discussed above in Sections 3.3.2 and 3.3.3, there is reasonable assurance that both the tube structural integrity and accident-induced leakage integrity performance criteria of TS 5.5.9.b.1 and TS 5.5.9.b.2, respectively, will be met for all tubes with existing known and potential degradation at the start of A2R22.

3.4 Technical Evaluation Conclusion

As discussed above, the TS requirements for SG tube integrity remain unaffected by the proposed changes and the licensee has demonstrated that SG tube integrity will be maintained with the one-time extension in frequency of the inspection. Thus, the NRC staff finds that the modified TS 5.5.9 SG inspection program provides administrative controls necessary to assure operation of the facility in a safe manner and continues to meet the requirements of 10 CFR 50.36(c)(5).

Based on the information submitted, the NRC staff finds that the licensee has demonstrated that there is reasonable assurance that the structural and leakage integrity of the Braidwood, Unit 2, SG tubes will be maintained until the next SG tube inspections during A2R22 in the fall of 2021. Based on the above, the NRC staff concludes that 10 CFR 50, Appendix A, GDCs 14, 15, and 31 are met. Therefore, the NRC staff concludes that the proposed changes to TS 5.5.9.d.3 described above are acceptable.

This NRC staff evaluation of the proposed one-time TS changes was performed within the context of the COVID-19 pandemic and the potential impacts of this virus to plant personnel safety and the potential for further spread of the virus to the public. 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.

In addition, the NRC staff finds that the proposed repagination of TS pages 5.5 – 8, 5.5 – 9, 5.5 – 10, and 5.5 – 11 is an editorial or format change that does not alter substantive content of TS 5.5.9. Therefore, the NRC staff finds the repagination acceptable.

4.0 EXIGENT CIRCUMSTANCES

Background

The NRC's regulations contain provisions for issuance of amendments when the usual 30-day public comment period cannot be met. These provisions are applicable under exigent circumstances. As stated in 10 CFR 50.91(a)(6), exigent circumstances exist when: (1) a licensee and the NRC must act quickly; (2) time does not permit the NRC to publish a *Federal Register* notice allowing 30 days for prior public comment; (3) the NRC determines that the amendment involves no significant hazards consideration, and (4) the licensee has made its best efforts to make a timely application. In its application dated April 6, 2020, as supplemented by letters dated April 13, 2020, and April 16, 2020, the licensee requested that the proposed amendment be processed by the NRC on an emergency basis; however, the NRC staff determined that exigent circumstances exist under 10 CFR 50.91(a)(6).

Under the provisions in 10 CFR 50.91(a)(6), the NRC notifies the public in one of two ways: (1) by issuing a *Federal Register* notice providing an opportunity for hearing and allowing at least 2 weeks from the date of the notice for prior public comments or (2) by using local media to provide reasonable notice to the public in the area surrounding the licensee's facility. In this case, the NRC provided notice in the *Federal Register* on April 16, 2020 (85 FR 21274).

The licensee provided the following information to support its need for this exigent LAR. The licensee indicated that the nature of the SG inspections conflicts with the recommendations in that they require workers to be in constant proximity to each other in a hot and radiological environment that increases the likelihood of individuals contracting COVID-19 and potentially inducing a rapid spread. Additionally, these inspections require a specialty vendor that maintains unique and complex qualifications. Losing resources due to a virus spread would cause a situation where the proper technical knowledge would not be available to satisfactorily complete this work because of the minimal 14-day isolation and the likelihood that more than one individual could be affected (based on having to work in close proximity for the work). Because the Braidwood, Unit 2, TSs require inspections be completed to determine operability of the SGs prior to entering Mode 4, which is planned for May 4, 2020, the NRC staff determined that the need for this amendment does not allow for the standard public comment period, but does allow for a 14-day comment period.

NRC Staff Conclusion

Based on the above circumstances, the NRC staff finds that the licensee and Commission need to act quickly, and time does not permit a 30-day prior comment period because the plant would be prevented from resuming operations following the refueling outage occurring after the current operating cycle that ends in April 2020. Therefore, the NRC staff provided a 14-day comment period. In addition, the NRC staff finds that the licensee made a timely application for the proposed amendment following identification of the issue because the licensee could not have anticipated the social distancing recommendations associated with the public health emergency caused by the COVID–19 virus and used best efforts to promptly submit its application for the proposed amendment. Based on these findings and the determination that the amendment involves no significant hazards consideration as discussed below, the NRC staff has determined that a valid need exists for issuance of the license amendment using the exigent provisions of 10 CFR 50.91(a)(6).

5.0 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 Steam Generator (SG) inspection to be performed after three operating cycles. This change does not physically change the SGs, the plant or the way the SGs or plant are operated. This change does not change the design of the SG. Inspection frequencies and inspection activities are not an initiator to a Steam Generator 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 Steam Generator (SG) inspection to be performed after three operating cycles. The proposed change does not alter the design function or operation of the SGs or the ability of an SG to perform the 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 th[a]n 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 Steam Generator (SG) inspection to be performed [until] after three 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 [Updated Final Safety Analysis Report] 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 proposed repagination of the identified TS pages 5.5 – 8 through 5.5 – 11 is an editorial and non-substantive format 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.

6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Illinois State official was notified of the proposed issuance of the amendment on April 16, 2020. The State official had no comments.

7.0 ENVIRONMENTAL CONSIDERATION

The issuance of an amendment revising TS 5.5.9.d.3 changes 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 published in the *Federal Register* on April 16, 2020 (85 FR 21274). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). The repagination of TS pages 5.5 – 8 through 5.5 – 11 is a minor editorial or format change to the renewed facility license that meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(10)(v). 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.

8.0 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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

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Date of Issuance: May 1, 2020

SUBJECT: BRAIDWOOD STATION, UNIT 2, - ISSUANCE OF AMENDMENT NO. 209 RE: ONE-TIME EXTENSION OF STEAM GENERATOR INSPECTIONS (EPID L-2020-LLA-0069) DATED MAY 1, 2020

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