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

September 21, 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: BYRON STATION, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENT NOS. 222 AND 222 RE: ONE-TIME EXTENSION OF UNIT NO. 2 STEAM GENERATOR INSPECTIONS [COVID-19] (EPID L-2020-LLA-0156)

Dear Mr. Hanson:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 222 to Renewed Facility Operating License No. NPF-37 and Amendment No. 222 to Renewed Facility Operating License No. NPF-66 for Byron Station, Unit Nos. 1 and 2, respectively. The amendments are in response to your application dated July 10, 2020 (Agencywide Documents Access and Management System (ADAMS) Package Accession No. ML20196L732), as supplemented by letter dated August 19, 2020 (ADAMS Accession No. ML20232D036).

The amendment to Byron Station, Unit No. 1, changes the amendment number to 222 because it has common technical specifications (TSs) with Byron Station, Unit No. 2. The amendment to Byron Station, Unit No. 2, revises TS 5.5.9, “Steam Generator (SG) Program," for a one-time revision to the frequency for SG tube inspections. The amendment allows deferral of the required inspections until the next Byron Station, Unit No. 2, refueling outage. In addition, TS pages 5.5 – 8, 5.5 – 9, and 5.5 – 10 are 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 Nos. STN 50-454,
and STN 50-455

Enclosures:
1. Amendment No. 222 to NPF-37
2. Amendment No. 222 to NPF-66
3. Safety Evaluation

cc: Listserv

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

EXELON GENERATION COMPANY, LLC

DOCKET NO. STN 50-454

BYRON STATION, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 222
Renewed License No. NPF-37

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

   A. The application for amendment by Exelon Generation Company, LLC (the licensee) dated July 10, 2020, as supplemented by letter dated August 19, 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 license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-37 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A as revised through Amendment No. 222 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this renewed license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 5 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert F. Kuntz

Digitally signed by Robert F. Kuntz
Date: 2020.09.21 09:01:39 -04'00'

Nancy L. Salgado, Chief (Robert Kuntz for)
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: September 21, 2020

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

EXELON GENERATION COMPANY, LLC

DOCKET NO. STN 50-455

BYRON STATION, UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 222
Renewed License No. NPF-66

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

   A. The application for amendment by Exelon Generation Company, LLC (the licensee) dated July 10, 2020, as supplemented by letter dated August 19, 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 license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-66 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A (NUREG-1113), as revised through Amendment No. 222, and the Environmental Protection Plan contained in Appendix B, both of which were attached to Renewed License No. NPF-37, dated November 19, 2015, are hereby incorporated into this renewed license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 5 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert F. Kuntz

Digitally signed by Robert F. Kuntz
Date: 2020.09.21 09:02:04 -04'00'

Nancy L. Salgado, Chief (Robert Kuntz for)
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: September 21, 2020

ATTACHMENT TO LICENSE AMENDMENT NOS. 222 AND 222

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

BYRON STATION, UNIT NOS. 1 AND 2

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

Replace the following pages of the Renewed Facility Operating Licenses and the Appendix A, Technical Specifications (TSs), 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-37	License No. NPF-37
Page 3	Page 3
License No. NPF-66	License No. NPF-66
Page 3	Page 3
TSs	TSs
Page 5.5 – 8	Page 5.5 – 8
Page 5.5 – 9	Page 5.5 – 9
Page 5.5 – 10	Page 5.5 – 10

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

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

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

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

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

(1) 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. 222 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this renewed license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Deleted.

(4) Deleted.

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

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

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

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

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

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

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

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

The Technical Specifications contained in Appendix A (NUREG-1113), as revised through Amendment No. 222, and the Environmental Protection Plan contained in Appendix B, both of which were attached to Renewed License No. NPF-37, dated November 19, 2015, are hereby incorporated into this renewed license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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 Programs and Manuals

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 B2R23 following inspections completed in refueling outage B2R20. 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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 222 TO RENEWED

FACILITY OPERATING LICENSE NO. NPF-37, AND AMENDMENT NO. 222

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-66

EXELON GENERATION COMPANY, LLC

BYRON STATION, UNIT NOS. 1 AND 2

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

1.0 INTRODUCTION

By application dated July 10, 2020 (Reference 1), as supplemented by letter dated August 19, 2020 (Reference 2), Exelon Generation Company, LLC (the licensee) requested a onetime change to the technical specifications (TSs) for Byron Station, Unit No. 2 (Byron, Unit No. 2). The proposed change would allow a onetime deferral of the steam generator (SG) tube inspections required in paragraph d.3 of TS 5.5.9 "Steam Generator (SG) Program," for Byron, Unit No. 2, until its next outage. The amendment for Byron Station, Unit No. 1, only changes the amendment number to 222 because it has common TSs with Byron Station, Unit No. 2. The proposed changes were submitted in response to social distancing recommendations by the United States Centers for Disease Control and Prevention, which were issued to prevent the spread of the Coronavirus Disease 2019 (COVID-19).[1]

The supplement dated August 19, 2020, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC or the Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on August 19, 2020 (85 FR 51078).

2.0 REGULATORY EVAULATION

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 (SE), 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.

[1] CDC, "How to Protect Yourself and Others," April 18, 2020 (ADAMS Accession No. ML20125A069).

2.2 Regulatory Requirements and Guidance

Requirements for the integrity of the SG tubing are established in Title 10 of the *Code of Federal Regulations* (10 CFR). Specifically, the general design criterion (GDC) 14, "Reactor coolant pressure boundary," in 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," and GDC 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 behave in a non-brittle manner. GDC 31 also requires that the probability of a rapidly propagating fracture be minimized.

The regulations in10 CFR 50.36, "Technical specifications," require that each applicant for an operating license includes (in its application) proposed TSs and establishes the regulatory requirements for the content of TSs. The TSs 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 TSs as the Commission finds appropriate.

Given the importance of SG tube integrity, all current pressurized-water reactor (PWR) licensees have TSs governing the surveillance of SG tubes. The TSs for all PWR plants require that an SG program be established and implemented to ensure that SG tube integrity is maintained. Programs established by the licensee, including the SG program, are listed in the administrative controls section of the TSs. For Byron, Unit No. 2, 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.

For Byron, Unit No. 2, SG tube integrity is maintained by meeting the performance criteria specified in TS Section 5.5.9.b for structural integrity and leakage, consistent with the plant design and licensing basis. TS Section 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 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 criteria, specified in TS Section 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 inservice through application of alternate repair criteria provided in TS Section 5.5.9.c.

Byron, Unit No. 2, TS Section 3.4.13, includes a limit on operational primary-to-secondary leakage, beyond which the plant must be promptly shut down. Should an existing flaw that exceeds the SG tube integrity repair limit not be detected during the periodic tube surveillance required by the plant TSs, the operational leakage limit provides added assurance of timely plant shutdown before SG tube integrity is impaired.

In the Byron Station Updated Safety Analysis Report (Reference 12), the licensee analyzed the consequences of a SG tube rupture (Section 15.6.3) and a steam line break (Section 15.1.5.3). 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 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 Byron, Unit No. 2 are being changed because of 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 DBAs for SG tubes.

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. Reference 1, as supplemented by Reference 2, proposed changes to TS 5.5.9, which provides plant-specific details on the Byron, Unit No. 2, SG inspection program. The NRC staff reviewed Reference 1 and 2 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 SG Design

Byron, Unit No. 2 has four Westinghouse Model D5 SGs, which have been in service since the plant began commercial operation in 1987. Each SG contains 4,570 thermally treated Alloy 600 tubes (Alloy 600TT) with an outside diameter of 0.750 inches and a nominal wall thickness of 0.043 inches. The tubes were hydraulically expanded to the full length of the tube sheet. The tubes are supported by stainless steel tube support plates (TSPs) with quatrefoil broached-holes and chrome-plated Alloy 600 anti-vibration bars (AVBs).

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. To limit tube vibration, 144 tubes were hydraulically expanded into the drilled holes in the second and third cold-leg baffle plates (02C and 03C, respectively) (Reference 3).

3.1.2 Operating Experience

All four SGs in Byron, Unit No. 2 were last inspected in fall 2014 (Refueling Outage 18 (B2R18)) and fall 2017 (B2R20). Additional information regarding the SG inspections at Byron, Unit No. 2, including the eddy current examinations performed, is available in the fall 2014 and fall 2017 SG Tube Inspection Reports (References 4 and 5, respectively).

Since the SGs were placed in-service in 1987, a total of 509 tubes have been plugged (159 in SG 2A, 142 in SG 2B, 166 in SG 2C, and 42 in SG 2D). Table 2, “[Byron, Unit No. 2] Plugging History by Degradation Mechanism,” of Attachment 1 to Reference 1 identifies the reasons 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). Byron, Unit No. 2, has 39 tubes identified as potentially having high residual stress in service, however, no SCC has been detected.

Byron, Unit No. 2, SGs have the following existing degradation mechanisms: mechanical wear from AVBs, mechanical wear from TSP quatrefoil, mechanical wear from drilled holes in preheater baffle plates (also called drilled support plates or DSPs), and mechanical wear from foreign objects (FOs). Inspections during B2R20 reported a total of 1,193 wear indications from all mechanisms in the four Byron, Unit No. 2, SGs. Table 1 summarizes the wear indications by location reported from B2R20, the latest inspection.

Table 1: Wear Indications Reported in Byron, Unit No. 2, SGs (B2R20, fall 2017)

Wear Location	**Total Number of Indications in Each SG**			
	SG 2A	**SG 2B**	**SG 2C**	**SG 2D**
AVB	259	402	293	191
TSP Quatrefoil	0	5	5	4
DSPs	0	2	1	2
FOs (New and Historical)	8	12	3	6

Secondary-side inspections in 2014 (B2R18) included: a full foreign object search and retrieval (FOSAR) of the preheater high-flow regions, including the preheater waterbox rib and cap plate region, on the preheater tube support baffle plate (TSP 02C) in SGs 2A and 2D; a visual inspection of the 2B SG upper bundle, including the tube lanes and four in-bundle columns at each TSP; visual and ultrasonic inspections of the primary and secondary-side moisture separator regions in SGs 2A and 2D; and a visual inspection of all four SG primary channel heads. No evidence of degradation or anomalies was reported in the 2A and 2D SG preheaters. Other than deposits comparable to those previously noted in the quatrefoils no other anomalous structural conditions, foreign objects, or erosion was observed in the visual inspections of the Upper Bundle in the 2B SG. Erosion wear was present and progressing slowly in the moisture separator region. No evidence of cladding or weld breaches or evidence of channel head wastage was identified in the SG primary channel heads. In addition, the licensee performed secondary-side top of tubesheet (TTS) sludge lancing in all four SGs. Secondary-side inspections in 2017 (B2R20) included: a full FOSAR of the preheater high-flow regions, including preheater waterbox, rib and cap plate region, on the pre-heater tube support baffle plate (TSP 02C) in SGs 2B and 2C; visual and ultrasonic inspections of the primary and secondary-side moisture separator regions in SGs 2A and 2C; and a visual inspection of all four SG primary channel heads. In addition, the licensee performed secondary-side chemical cleaning and TTS sludge lancing in all four SGs. New indications of FO wear on eight tubes were identified in 2017. Two tubes with FO wear indications were preventatively plugged in 2017 because an FO was assumed to be present at each wear location (see Section 3.3.2, "Evaluation of Existing Tube Degradation Mechanisms," of this SE for additional information regarding evaluation of FO wear).

In 2017, the licensee found and removed a backing bar in the preheater region of SG 2C. The licensee determined the source of the loose part to be one of the two cutouts made in the central location of the waterbox's cap plate region. These cutouts were made in the waterbox cap plate to access the inside of the waterbox. The cutout plates were reinstalled in the cap plate by full-penetration groove welds on three sides of each cut-out plate. The licensee used three backing bars for each plate to support the three sides of each cutout plate during welding. To account for the postulated event that the backing bars or cutout plates become loose parts, the licensee plugged and stabilized 91 tubes in SG 2C to prevent the potential loose parts from contacting and damaging active tubes. In addition, the licensee performed an extent of condition assessment for the remaining SG preheaters. In 2004 (B2R11), the licensee found and removed two carbon steel backing bars in the preheater region of SG 2A (Reference 11). The licensee also preventatively plugged and stabilized 91 tubes in SG 2A. In Reference 1, the licensee states that the SG 2A waterbox cap plate was repaired in 2005. The SG 2B waterbox cap plate has not been modified and, therefore, no backing bars are present and the SG 2D preheater has no backing bars.

TS Section 5.6.9.h requires the licensee to include leakage greater than 3 gallons per day in the Steam Generator Tube Inspection Report. The licensee stated that no primary-to-secondary leakage was noted during the past two operating cycles (21 and 22). The licensee explained this further by stating that all trends are below 3 gallons per day.

3.2 Proposed TS Changes

3.2.1 Current TS Requirements

Byron, Unit No. 2, TS Section 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.

Byron, Unit No. 2, TS Sections 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 Section 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 Section 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." Byron, Unit No. 2, is currently in the fourth inspection period; therefore, the TSs require 100 percent of the tubes to be inspected every 72 effective full power months.

3.2.2 Description of Proposed TS Changes

Reference 1, as supplemented by Reference 2, proposes to revise TS Section 5.5.9.d.3 to add a onetime exception to the maximum inspection interval of 48 months or every other refueling outage. The exception would require inspection at the third refueling outage (B2R23) after the last inspection (B2R20). TS Section 5.5.9.d.3 currently would require inspection during refueling outage 22 (B2R22), which begins in October 2020. B2R23 will occur approximately 18 months later, in the spring of 2022. The proposed revised part of TS Section 5.5.9.d.3 reads as follows, with the underlined portion being added for the onetime exception:

> 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)<u> with the exception that each SG is to be inspected during the third refueling outage in B2R23 following inspections completed in refueling outage B2R20</u>. 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.

The above change also results in the repagination of TS pages 5.5 – 8 through 5.5 – 10 and the incrementing of the Byron, Unit No. 1, amendment number, which are editorial or format changes.

3.3 Staff Evaluation of Proposed TS Changes

3.3.1 Evaluation Approach

The NRC staff evaluation of the proposed change to the TS required SG inspection frequency 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. That said, in granting this license amendment, the NRC staff still finds reasonable assurance of the public health and safety. The NRC staff evaluated the proposed change 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 changes to its current TS limiting conditions for operation and surveillance requirements, and states in its application: "The proposed one-time revision to TS 5.5.9 to defer the SG inspection . . . does not alter Byron Unit [No.] 2 compliance with the referenced TS LCOs or the requirements of 10 CFR 50.36."

The NRC staff evaluation of the proposed TS changes focused on the potential to affect SG tube integrity, since maintaining SG tube integrity ensures the plant will meet its SG Program TS, thereby protecting the public's health and safety. In particular, the staff evaluation assessed whether the licensee's requested amendment demonstrates that the structural integrity performance criterion (SIPC) and accident-induced leakage performance criterion (AILPC) will be met for cycle 23 until the spring 2022 refueling outage (B2R23). These criteria are defined in TS Section 5.5.9.b.

The Byron, Unit No. 2, SGs have experienced tube degradation from wear against support structures and FOs; these degradation mechanisms are classified as existing mechanisms. The Byron, Unit No. 2, SGs have not experienced SCC mechanisms, meaning these mechanisms are classified as potential mechanisms. In Reference 1, the licensee provided an analysis of existing wear mechanisms in a deterministic manner while potential SCC mechanisms were analyzed in a full-bundle probabilistic manner, as outlined in the Electric Power Research Institute (EPRI) Integrity Assessment Guidelines (IAGL) (Reference 8). 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 B2R20 inspections for AVB wear consisted of full-length bobbin probe examinations of 100 percent of the in-service tubes. There were 1,145 AVB wear indications detected during B2R20. The licensee, in Reference 1, Attachment 6, Section 5.2, stated that the AVB wear indications ranged in depth from 9 to 42 percent through-wall (TW). The AVB wear indications were sized using an EPRI-qualified examination technique. Two indications in SG 2A that exceeded the TS plugging limit of 40 percent TW were plugged during B2R20. The licensee, in Reference 1, Attachment 6, Section 5.2, states that the largest AVB wear indications allowed to remain in service in B2R20 were 39 percent TW; one each in SG 2A and 2B. The licensee applied the upper 95th percentile AVB wear rate of 2.1 percent TW per effective full power year (EFPY) and projected the wear depth at B2R23 (after three operating cycles) to be 48 percent TW, which is less than the CM limit of 68 percent TW (determined by 3 times normal operating pressure differential (3xNOPD)). The SIPC in TS 5.5.9.b requires a safety factor of 3 against burst at NOPD. This is satisfied if the wear rate is less than the CM limit. Therefore, the SIPC will be satisfied. The AILPC in TS 5.5.9.b 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 gallon per minute (gpm) for all SGs. Byron, Unit No. 2, will maintain structural integrity so long as it maintains the 3.0 margin against burst under normal steady state full-power operation primary-to-secondary pressure differential. This means that as long as Byron, Unit No. 2, maintains that 3.0 margin, its projected accident tube leakage will be zero and the AILPC will be satisfied. Therefore, based on the limiting depth size of 48 percent TW for AVB wear at B2R23, which maintains that 3.0 margin, Byron, Unit No. 2's cumulative projected accident tube leakage will be zero. Based on zero leakage, the NRC staff determined that the AILPC is satisfied.

<u>Wear at Tube Support Plates</u>

The B2R20 inspections for TSP wear consisted of full-length bobbin probe examinations of 100 percent of the in-service tubes. There were 14 TSP wear indications detected during B2R20 that ranged in depth from 11 to 39 percent TW. The TSP wear indications were sized using an EPRI qualified examination technique. One tube in SG 2C was plugged due to TSP wear during B2R20. The largest TSB wear indication allowed to remain in service in B2R20 was 34 percent TW in SG 2C. The licensee, in Reference 1, states that due to the small population of TSP wear indications in the SGs, there was not sufficient data to develop accurate growth rate curves, so, instead of the 95th percentile growth rate, analyses of TSP wear applied the maximum observed growth rate to the largest flaw left in service during the last inspection. The maximum observed growth rate of 3.2 percent TW per EFPY was applied and the projected wear depth at B2R23 (after three operating cycles) was 48 percent TW, which is less than the CM limit of 52 percent TW (determined by 3xNOPD). Since the factor of 3 margin is maintained, as discussed above, by the 48 percent TW projection, the NRC staff determined that the SIPC is satisfied. Further, as discussed above, Byron, Unit No. 2, will satisfy the AILPC so long as it maintains the margin of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential. Therefore, based on the limiting depth size of 48 percent TW for TSP wear at B2R23, which maintains that 3.0 margin, Byron, Unit No. 2's, cumulative projected accident tube leakage will be zero. Based on zero leakage, the NRC staff determined that the AILPC is satisfied.

<u>Wear at Drilled Support Plates</u>

Drilled support plates refer to the lowermost flow baffle plate (01H and 01C) and the drilled hole support plates in the preheater section that range between 06C and 02C, inclusive. The B2R20 inspections for DSP wear consisted of full-length bobbin probe examinations of 100 percent of the in-service tubes. There were five DSP wear indications detected during B2R20 that ranged in depth from 9 to 26 percent TW. The DSP wear indications were sized using an EPRI qualified examination technique using a +Point™ probe. No tubes were plugged due to DSP wear during B2R20. The largest DSP wear indication allowed to remain in service in B2R20 was 26 percent TW in SG 2C. The licensee, in Reference 1, states that due to the small population of DSP wear indications in the SGs, there was not sufficient data to develop accurate growth rate curves, so, instead of the 95th percentile growth rate, analyses of DSP wear applied the maximum observed growth rate to the largest flaw left in-service during the last inspection. The largest observed wear rate was 7.1 percent TW per EFPY from B2R18 to B2R20 on one tube in SG 2C and this wear was considered atypical by the licensee for Byron, Unit No. 2. The licensee, in Reference 1, Attachment 6, Section 5.4, explained that in B2R18, the DSP indications were sized using bobbin probe and in B2R20 they were sized using +Point™ probe. The licensee reviewed the bobbin data for this location and determined that this was not a valid DSP growth rate because there was no change in the bobbin signals between the B2R18 and B2R20 inspections, which indicates no growth. Therefore, the maximum observed growth rate was found on two tubes, one each in SGs 2B and 2D as shown in Reference 1, Attachment 6, Section 5.4. As stated by the licensee in Reference 1, Attachment 6, Section 5.4, it applied the maximum observed growth rate of 2.5 percent TW per EFPY and found that the projected wear depth at B2R23 (after three operating cycles) was 37 percent TW, which is less than the CM limit of 53 percent TW (determined by 3xNOPD). Since the factor of 3 margin is maintained, as discussed above, by the 37 percent TW projection, the NRC staff determined that the SIPC is satisfied. Further, as discussed above, Byron, Unit No. 2, will satisfy the AILPC so long as it maintains the margin of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential. Therefore, based on the limiting depth size of 37 percent TW for DSP wear at B2R23, which maintains that 3.0 margin, Byron, Unit No. 2's, cumulative projected accident tube leakage will be zero. Based on zero leakage, the NRC staff determined that the AILPC is satisfied.

<u>Evaluation Summary for Wear at AVBs, TSPs, and DSPs</u>

Wear at these locations has been effectively managed for many cycles without challenging tube integrity. During SG tube inspections, the licensee performs CM to assess whether the measured flaw sizes are bounded by the previous operational assessment (OA) flaw size projections. During the most recent Byron, Unit No. 2, inspection in B2R20, the OA worst case projections from B2R18 were bounding for the measured tube wear mechanisms in B2R20, providing confidence that the OA methods and input assumptions can conservatively project future performance. The projections of the new OA submitted in Reference 1 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 evaluation of wear at AVBs, TSPs, and DSPs to be acceptable.

<u>Foreign Object Wear</u>

In addition to wear at support structures, Byron, Unit No. 2, has also experienced SG tube wear from FOs that have been transported into the SGs. At the conclusion of B2R20, 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 (Attachment 11 to Reference 1). The assessment for a third cycle of operation addressed the FOs identified during the B2R20 FOSAR, as well as the newly reported volumetric wear indications in the upper tube bundle, FOs and upper bundle wear indications remaining in service from previous operating cycles, and the possibility of waterbox backing bars in SG 2C to cause tube wear. The licensee identified eight tubes with new indications of FO wear in 2017. FOs were determined not to be present for six of the FO wear indications. However, two tubes with FO wear indications were preventatively plugged in 2017 because an FO was assumed to be present at each wear location. The FOs were not visually confirmed due to the limited accessibility of the two tubes. Based on +Point™ probe eddy current data, a FO is positioned within the quatrefoil flow hole of each of the two plugged tubes and the FOs did not contact adjacent tubes. In addition, the licensee stated that there is no risk of the two plugged tubes severing and affecting adjacent tubes since the circumferential extent of the wear indications is limited to less than 90 degrees. Therefore, the licensee concluded that FO wear of in-service tubes sufficient to degrade tube integrity to below acceptance criteria from these remaining FOs is not expected during the time up to B2R23. The NRC staff determined that based on the above discussion, the licensee demonstrated that SG tube integrity will not be impacted by the identified wear from FOs up to B2R23.

The potential for waterbox backing bars to become FOs is discussed in Section 3.1.2 of this SE. Based on that discussion, the NRC staff determined that the licensee's actions regarding the waterbox backing bars are sufficient to prevent additional FOs from this source.

During the B2R20 steam drum inspections, 6 of 32 spacer tabs in SG 2A and 9 of 32 spacer tabs in SG 2C were found to have degradation, but no through-wall holes. The licensee stated that the as-found condition of the spacer tabs met the thickness acceptance criteria and that the spacer tabs have been in operation since plant startup. Based on this, the licensee does not expect the spacer tabs to degrade enough to cause loose parts for one additional cycle. The NRC staff reviewed the licensee's analysis and conclusion and found it reasonable.

The NRC staff determined that the licensee's evaluation of FOs is acceptable since it accounts for tube wear from known FOs within the SGs. Based on this analysis, the NRC staff determined that the licensee demonstrated that tube integrity will be maintained until B2R23.

3.3.3 Evaluation of Potential Tube Degradation Mechanisms

In addition to existing tube degradation mechanisms, the licensee considered potential tube degradation mechanisms. The NRC staff previously approved an H* [2] amendment for Byron, Unit No. 2 (Reference 9), concluding in the related SE that potential tube degradation below the H* depth in the tubesheet does not affect tube integrity. Therefore, the licensee's evaluation of potential mechanisms in this SE does not consider potential tube degradation between the H* distance and the tube end.

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 the 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 temperatures. Similarly, hotter portions of the tubing are generally more susceptible to SCC than colder sections of the tubing. Higher residual stress is also an accelerating factor for SCC in SG tubing. SGs with Alloy 600TT tubes are known to have some tubes with higher residual stress that are more susceptible to cracking. Byron, Unit No. 2, has 39 tubes identified as potentially having high residual stress in service, however, no SCC has been detected.

Although SCC has not been detected in the Byron, Unit No. 2, SG tubing, eddy current examinations with specialized probes are performed at refueling outages to detect cracking if it

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

forms. The licensee performed the following eddy current inspections in B2R20 to detect potential forms of SCC not previously identified at Byron, Unit No. 2:

- Axial ODSCC at TSP intersections on known high residual stress tube locations: All 39 tubes identified as potentially having high residual stress received a full-length bobbin probe inspection and X-Probe™ inspection of all hot- and cold-leg TSP intersections.

- Axial ODSCC at tubing dents and dings: The licensee conducted full-length bobbin probe inspections of 100 percent of in-service tubes for detection of axial ODSCC at tubing dents and dings up to 5 volts. Also, the licensee inspected a 50 percent sample of dents and dings greater than 5 volts in the hot-leg, U-bend, and outside the preheater with a +Point™ probe. Additionally, Byron, Unit No. 2, inspected a 50 percent sample of dents and dings greater than 2 volts in the preheater and flow distribution baffle on the hot-leg with a +Point™ probe (the remaining 50 percent were inspected in B2R18). In addition, in both B2R18 and B2R20, the licensee tested all dents and dings greater than 2 volts in tubes identified as potentially having high residual stress a +Point™ probe.

- Circumferential and Axial ODSCC and PWSCC at the hot-leg TTS expansion transition (including over-expansions) and inside the tubesheet (including bulges) to the H* depth: The licensee inspected 50 percent of the bundle plus 2–3 tubes in the periphery with the X-Probe™. The licensee inspected the remainder in B2R18.

- Axial PWSCC in small radius U-bends: The licensee inspected a 50 percent sample of the Row 1 and Row 2 U-bends with a +Point™ probe. The licensee inspected the remaining 50 percent in B2R18. In addition, all tubes with the "Blairsville Bump" manufacturing anomaly were inspected with a +Point™ probe in B2R18 and B2R20. More information about the “Blairsville Bump” is contained in Section 4.4.3 of Reference 3.

The OA for potential degradation mechanisms projects the behavior of postulated flaws that could have been present at, or prior to, the last SG inspection in B2R20 and those that could initiate during the 3-cycle operating period until B2R23. The OA for 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 (Reference 8**Error! Bookmark not defined.**) were applied for axial and circumferential ODSCC at hot-leg TTS expansion transitions and for axial ODSCC at freespan dents and dings in non-high-stress tubes.

- Axial ODSCC at TSP intersections on non-high residual stress tubes was reported for the first time in the A600TT fleet in fall 2019. This mechanism was addressed by the axial ODSCC at TSP intersections on high residual stress tubes with a low Weibull slope initiation model. This analysis applied the EPRI IAGL (Reference 8) upper bound default growth rates.

- Axial and circumferential PWSCC at TTS expansion transitions are bounded by the developed axial and circumferential ODSCC growth rates, which are bounded by the EPRI IAGL (Reference 8) typical default growth rates.

- For SCC mechanisms that were sampled in B2R20, except for axial ODSCC at dents and dings, the tube population was divided into two groups per the implemented sampling plan (inspected and non-inspected) in accordance with the EPRI IAGL (Reference 8). 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.

- The assumed first initiation point for axial ODSCC at dents and dings was assumed earlier in the plant operating history than for other SCC mechanisms, due to detection challenges at these locations.

The 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 the EPRI IAGL (Reference 8). 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 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 NRC staff considers the probabilistic evaluation assumptions to be conservative for Byron, Unit No. 2. Although no SCC has been detected at Byron, Unit No. 2, the model assumes that forms of unidentified SCC existed at the previous inspection, and the cracks will grow until the B2R23 inspections. These model assumptions were based on fleet-wide cracking experience with Alloy 600TT SG tubing. As the licensee states in Reference 1, Attachment 6, the model predicts that axial and circumferential cracks will be detected at the TTS (Sections 6.4 and 6.5, respectively), axial cracks will be detected at the TSP intersections (Section 6.3), and axial cracks will be detected at dents/dings (Section 6.6) at the B2R23 inspection. The calculated probability of burst for all potential mechanisms satisfy the SIPC safety factors until the B2R23 inspections. The accident-induced leakage was determined by summing the projected leak rates at B2R23. The licensee states in Reverence 1, Attachment 6, Section 6.1 that Byron, Unit No. 2, has not experienced any corrosion degradation of the pressure boundary portion of the SG tubing. However, the licensee's OA assumed that all such potential mechanisms were active and the summed leak rate for all potential mechanisms was determined to be 0.0757 gallons per minute. This leakage value is less than the AILPC leak limit of 0.5 gallons per minute.

Therefore, based on the evaluation discussed above in Sections 3.3.2 and 3.3.3, the NRC staff finds that there is reasonable assurance that both the structural integrity and accident-induced leakage performance criteria will be met for all SG tubes with existing known and potential degradation until the B2R23.

3.4 Main Steam Pressure and T-ave Impact

To recover main steam pressure prior to B2R23, the licensee may increase average operating temperature (T-ave) by up to 1 degree Fahrenheit. The licensee does not expect an increase in T-ave to adversely impact SCC initiation in the SG tubes because other Alloy 600TT units have operated at higher T-ave values than Byron, Unit No. 2. Because Byron, Unit No. 2, has typically operated around 611 degrees Fahrenheit , which is lower than other similar units which have not been adversely impacted by SCC initiation, and since SCC initiation is a temperature dependent phenomenon, the NRC staff finds that increasing T-ave by up to 1 degree Fahrenheit is not expected to adversely impact SCC initiation at Byron, Unit No. 2, by B2R23.

3.5 Mitigating Strategies

In Reference 1, the licensee states 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 gallons per day (gpd). The Exelon Generating Company 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 does not require this as a condition for acceptance of the TS change. It finds this acceptable, since this limit requires shutdown at a lower leakage level, which reduces the probability of a SG tube leak progressing to a SG tube burst.

3.6.1 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 Byron, Unit No. 2, SG tube integrity will be maintained with the onetime 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 integrity of the Byron, Unit No. 2, SG tubes will be maintained until the next SG tube inspections during the spring of 2022. 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 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 NRC 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, and 5.5 – 10 and the incrementing of the Byron, Unit No. 1, amendment number are editorial or format changes that do not alter substantive content of TS 5.5.9. Therefore, the NRC staff finds the repagination and the incrementing of the Byron, Unit No. 1, amendment number acceptable.

Based on the above, the NRC staff finds that the licensee has demonstrated that there is reasonable assurance that the structural and leakage integrity of the Byron, Unit No. 2 SG tubes will be maintained until the next SG tube inspections during B2R23 in the spring of 2022. Therefore, the NRC staff concludes that the proposed changes are acceptable.

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.

As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the proposed change 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 three 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 change 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 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 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 change 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 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 license amendment request (Reference 1, as supplemented by Reference 2) 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 – 10 and the incrementing of the Byron, Unit No. 1, amendment number are editorial and non-substantive format changes 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 amendments and that the amendments 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 September 2, 2020. The State official had no comments.

6.0 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 August 19, 2020 (85 FR 51078). Accordingly, the amendment for Byron, Unit No. 2, 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 – 10 and the incrementing of the Byron, Unit No. 1, amendment number are editorial or format changes to the renewed facility licenses that meet 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 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. Byron 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 July 10, 2020 (Agencywide Documents Access and Management System (ADAMS) Package Accession No. ML20196L732).

2. Byron 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 August 19, 2020 (ADAMS Accession No. ML20232D036).

3. NUREG-2188, "U.S. Operating Experience with Thermally Treated Alloy 600 Steam Generator Tubes Through December 2013," dated February 29, 2016 (ADAMS Accession No. ML16061A159).

4. Byron Station, Unit 2, Steam Generator Tube Inspection Report for Refueling Outage 18, dated February 20, 2015 (ADAMS Accession No. ML15051A312).

5. Byron, Unit 2, Steam Generator Tube Inspection Report for Refueling Outage 20, dated April 5, 2018 (ADAMS Accession No. ML18095A116).

6. IMC 0327, "Steam Generator Tube Primary-to-Secondary Leakage," dated January 1, 2019 (ADAMS Accession No. ML18093B067).

7. 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).

8. Steam Generator Management Program: Steam Generator Integrity Assessment Guidelines: Revision 4. EPRI, Palo Alto, CA: 2016. 3002007571.

9. Braidwood Station, Units 1 and 2 and Byron Station, Unit Nos. 1 and 2 – Issuance of Amendments Re: Revise Technical Specifications 5.5.9 and 5.6.9, For Permanent Alternative Repair Criteria, dated October 5, 2012 (ADAMS Accession No. ML12262A360).

10. Bryon, Unit 1, Current Facility Operating License NPF-37, Tech Specs, Revised 07/24/2020 (ADAMS Accession No. ML052910365). (Note: Bryon Units 1 and 2 share TS).

11. Summary of Conference Call with Exelon Generation Company, LLC Regarding the Results of the Spring 2004 Steam Generator Inspections at Byron Station, Unit 2, August 12, 2004 (ADAMS Package Accession No. ML042260202).

12. Braidwood Station, Units 1 and 2, and Byron Station, Unit Nos. 1 and 2, Updated Final Analysis Report, Chapter 15, Revision 17, December 17, 2018 (ADAMS Accession No. ML19170A347).

Primary Reviewers: A. Johnson
L. Terry

Date of issuance: September 21, 2020

SUBJECT: BYRON STATION, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENT NOS. 222 AND 222 RE: ONE-TIME EXTENSION OF UNIT NO. 2 STEAM GENERATOR INSPECTIONS [COVID-19] (EPID L-2020-LLA-0156) DATED SEPTEMBER 21, 2020

DISTRIBUTION:
PUBLIC
PM File Copy
RidsNrrDssStsb Resource
RidsRgn3MailCenter Resource
RidsNrrDorlLpl3 Resource
RidsNrrLASRohrer Resource
RidsNrrPMByron Resource
RidsNrrPMBraidwood Resource
RidsACRS_MailCTR Resource
RidsNrrDnrlNscg Resource
RidsNrrDssStsb Resource

ADAMS Accession No.: ML20246G865 ***e-mail concurrence**

OFFICE	NRR/DORL/LPL3/PM	NRR/DORL/LPL3/LA	NRR/DNRL/NCSG/BC*	NRR/DSS/STSB/BC*
NAME	JWiebe	SRohrer	SBloom	VCusumano*
DATE	9/4/20	9/3/20	8/30/20	9/9/20
OFFICE	OGC*	NRR/DORL/LPL3/BC*	NRR/DORL/LPL3/PM	
NAME	NMertz	NSalgado (R. Kuntz for)	JWiebe	
DATE	9/16/20	9/18/20	9/21/20	

OFFICIAL RECORD COPY