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

May 7, 2020

Mr. Daniel G. Stoddard
Senior Vice President and
Chief Nuclear Officer
Innsbrook Technical Center
5000 Dominion Blvd.
Glen Allen, VA 23060-6711

SUBJECT: SURRY POWER STATION, UNIT NOS. 1 AND 2, ISSUANCE OF EXIGENT AMENDMENT NOS. 299 AND 299 TO REVISE TECHNICAL SPECIFICATION 6.4.Q, "STEAM GENERATOR (SG) PROGRAM," TO ALLOW A ONE-TIME DEFERRAL OF THE SURRY UNIT NO. 2 SG "B" SPRING 2020 REFUELING OUTAGE INSPECTION (EPID NO. L-2019-LLA-0071)

Dear Mr. Stoddard:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 299 to Renewed Facility Operating License No. DPR-32 and Amendment No. 299 to Renewed Facility Operating License No. DPR-37 for the Surry Power Station (Surry), Unit Nos. 1 and 2, respectively. The amendments revise the Technical Specifications (TSs) in response to your application dated April 14, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20105A223).

The amendments revise the Surry, Unit Nos. 1 and 2, TS 6.4.Q.4.b to add a note to permit a one-time deferral of the Surry, Unit No. 2 Steam Generator "B" inspection from the spring 2020 refueling outage (RFO) (2R29) to the fall 2021 RFO (2R30).

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

Sincerely,

/RA/

G. Edward Miller, Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-280 and 50-281

Enclosures:
1. Amendment No. 299 to DPR-32
2. Amendment No. 299 to DPR-37
3. Safety Evaluation

cc: Listserv

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

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-280

SURRY POWER STATION, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 299
Renewed License No. DPR-32

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

   A. The application for amendment by Virginia Electric and Power Company (the licensee) dated April 14, 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 3.B of Renewed Facility Operating License No. DPR-32 is hereby amended to read as follows:

   B Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 299 are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley

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Date: 2020.05.07 13:44:19 -04'00'

Michael T. Markley, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to License No. DPR-32
and the Technical Specifications

Date of Issuance: May 7, 2020

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-281

SURRY POWER STATION, UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 299
Renewed License No. DPR-37

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

   A. The application for amendment by Virginia Electric and Power Company (the licensee) dated April 14, 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 3.B of Renewed Facility Operating License No. DPR-37 is hereby amended to read as follows:

   B Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 299 are hereby incorporated in this renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley

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Date: 2020.05.07 13:44:55 -04'00'

Michael T. Markley, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to License No. DPR-37
and the Technical Specifications

Date of Issuance: May 7, 2020

ATTACHMENT TO AMENDMENT NO. 299

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-32

AND AMENDMENT NO. 299

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-37

SURRY POWER STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-280 AND 50-281

Replace the following pages of the Licenses and the Appendix A Technical Specifications (TSs) with the attached revised pages. The revised pages are identified by amendment number and contained marginal lines indicating the areas of change.

Renewed Facility Operating License No. DPR-32

REMOVE	INSERT
3	3

Renewed Facility Operating License No. DPR-37

REMOVE	INSERT
3	3

TSs

REMOVE	INSERT
6.4-13	6.4-13

3. This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations: 10 CFR Part 20, Section 30.34 of 10 CFR Part 30, Section 40.41 of 10 CFR Part 40, Sections 50.54 and 50.59 of 10 CFR Part 50, and Section 70.32 of 10 CFR Part 70; and is subject to all applicable provisions of the Act and the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified below:

   A. Maximum Power Level

      The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2587 megawatts (thermal).

   B. Technical Specifications

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 299 are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

   C. Reports

      The licensee shall make certain reports in accordance with the requirements of the Technical Specifications.

   D. Records

      The licensee shall keep facility operating records in accordance with the requirements of the Technical Specifications.

   E. Deleted by Amendment 65

   F. Deleted by Amendment 71

   G. Deleted by Amendment 227

   H. Deleted by Amendment 227

   I. Fire Protection

      The licensee shall implement and maintain in effect the provisions of the approved fire protection program as described in the Updated Final Safety Analysis Report and as approved in the SER dated September 19, 1979, (and Supplements dated May 29, 1980, October 9, 1980, December 18, 1980, February 13, 1981, December 4, 1981, April 27, 1982, November 18, 1982, January 17, 1984, February 25, 1988, and

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

3. This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations: 10 CFR Part 20, Section 30.34 of 10 CFR Part 30, Section 40.41 of 10 CFR Part 40, Sections 50.54 and 50.59 of 10 CFR Part 50, and Section 70.32 of 10 CFR Part 70; and is subject to all applicable provisions of the Act and the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified below:

A. Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power Levels not in excess of 2587 megawatts (thermal)

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 299 are hereby incorporated in this renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

C. Reports

The licensee shall make certain reports in accordance with the requirements of the Technical Specifications.

D. Records

The licensee shall keep facility operating records in accordance with the Requirements of the Technical Specifications.

E. Deleted by Amendment 54

F. Deleted by Amendment 59 and Amendment 65

G. Deleted by Amendment 227

H. Deleted by Amendment 227

b. 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 b.1, b.2, and b.3 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.

1. 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;

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

3. 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.

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*As approved by Amendment Nos. 299 and 299, the inspection of Surry Unit 2 SG B may be deferred, on a one-time basis, from the Surry Unit 2 spring 2020 refueling outage (S2R29) to the Surry Unit 2 2021 fall refueling outage (S2R30).

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 299 TO

RENEWED FACILITY OPERATING LICENSE NO. DPR-32

AND AMENDMENT NO. 299 TO

RENEWED FACILITY OPERATING LICENSE NO. DPR-37

VIRGINIA ELECTRIC AND POWER COMPANY

SURRY POWER STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-280 AND 50-281

1.0 INTRODUCTION

By letter dated April 14, 2020 (Reference 1), Virginia Electric and Power Company (the licensee) requested changes to the Technical Specifications (TSs) for Surry Power Station (Surry), Unit Nos. 1 and 2. The proposed changes would allow a one-time deferral of the steam generator (SG) tube inspections required in TS 6.4.Q.4.b for Surry, Unit No. 2, SG "B" only from the spring 2020 refueling outage (RFO) (2R29) to the fall 2021 RFO (2R30).

The license amendment request (LAR) was submitted in response to social distancing recommendations by the Centers for Disease Control and Prevention, which have been issued as a defensive measure against the spread of the Coronavirus Disease 2019 (COVID-19). The licensee requested that the LAR be approved under exigent circumstances in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.91(a)(6).

2.0 REGULATORY EVALUATION

2.1 Description of System

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

2.2 Regulatory Requirements and Guidance

Fundamental regulatory requirements with respect to the integrity of the SG tubes are established in 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix A, "General Design Criteria for Nuclear Power Plants." Specifically, general design criterion (GDC) 14 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" and GDCs 15 and 31 state that the RCPB shall be designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including anticipated operational occurrences. In addition, GDC 30 states that the RCPB shall be "designed, fabricated, erected, and tested to the highest quality standards practical" and GDC 32 states that RCPB components shall be "designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity."

For plants that were issued construction permits before the effective date of 10 CFR Part 50, Appendix A, the plant-specific principal design criteria in the plant design basis established similar fundamental regulatory requirements pertaining to the integrity of the SG tubes. Surry, Unit No. 2 received a construction permit prior to May 21, 1971, which is the date that the GDC in Appendix A of 10 CFR Part 50 became effective. The Surry, Unit No. 2 Updated Final Safety Analysis Report (UFSAR) discusses the design criteria in effect at the time of construction in Section 1.4, while compliance with the criteria are discussed in Sections 4.1.2 and 4.1.3 of the UFSAR.

Section 182(a) of the Atomic Energy Act of 1954, as amended, requires nuclear power plant operating licenses to include TS. 10 CFR 50.36, "Technical specifications," contains the NRC regulatory requirements related to the content of the TS. Given the importance of SG tube integrity, all current pressurized-water reactor (PWR) licenses have TS governing the surveillance of SG tubes. The TSs require that an SG Program be established and implemented to ensure that SG tube integrity is maintained. Programs established by licensees, including the SG Program, are listed in the administrative controls section of the TS. For Surry, the requirements for performing SG tube inspections and plugging are in TS 6.4.Q, while the requirements for reporting the SG tube inspections and plugging are in TS 6.6.A.3. The NRC staff applied the following TS regulatory requirements for review of the LAR:

- Section 10 CFR 50.36, "Technical specifications," requires, in part, that the TSs and a summary statement of the bases or reasons for such specifications, shall be included by applicants for a license authorizing operation of a production or utilization facility. Specifically, 10 CFR 50.36(c) requires that TSs include items in the following categories: (1) safety limits, limiting safety system settings, and limiting control settings, (2) limiting conditions for operation (LCOs), (3) surveillance requirements, (4) design features, and (5) administrative controls. The proposed changes of these amendments relate to the LCO category.

- Section 10 CFR 50.36(c)(2), "Limiting conditions for operation," provides the requirements for the establishment of TS LCOs. Specifically, 10 CFR 50.36(c)(2)(i) provides that LCOs are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When an LCO is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the TSs until the condition can be met.

- Section 10 CFR 50.36(c)(3), "Surveillance requirements," provides the requirements for the establishment of TS SRs. SRs are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the LCOs will be met.

- Section 10 CFR 50.36(c)(5), “Administrative controls,” provides the requirements for the establishment of TS administrative controls. 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. Each licensee shall submit any reports to the Commission pursuant to approved technical specifications as specified in 10 CFR 50.4.

For Surry, SG tube integrity is maintained by meeting the performance criteria specified in TS 6.4.Q.2 for structural and leakage integrity, consistent with the plant design and licensing basis. TS 6.4.Q.1 requires that a condition monitoring assessment be performed during each outage in which the SG tubes are inspected, to confirm that the performance criteria are being met. TS 6.4.Q.4 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 6.4.Q.3, are that tubes found during in-service inspection to contain flaws with a depth equal to or exceeding 40 percent of the nominal wall thickness shall be plugged, unless the tubes are permitted to remain in service through the application of alternate tube plugging criteria provided in TS 6.4.Q.3.a.

Surry TS 3.1.C includes a limit on operational primary-to-secondary leakage, beyond which the plant must be promptly shutdown. Should a flaw exceeding the SG tube plugging limit not be detected during the periodic tube surveillance required by the TS, the operational leakage limit provides added assurance of timely plant shutdown before tube structural and leakage integrity are impaired, consistent with the design and licensing basis.

As part of the plant’s licensing basis, applicants for PWR licenses are required to analyze the consequences of postulated design-basis accidents (DBAs), such as an 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 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 Surry are being changed because of the proposed amendments 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.

3.0 TECHNICAL EVALUATION

3.1 Background

3.1.1 Steam Generator Design

Surry, Unit No. 2 has three Westinghouse model 51F SGs. Each SG has 3,342 thermally-treated Alloy 600 (Alloy 600TT) tubes with a nominal outside diameter of 0.875 inches and a nominal wall thickness of 0.050 inches. The tubes are hydraulically expanded at each end for the full depth of the tubesheet. The tubes are supported by stainless steel tube support plates (TSPs) with quatrefoil-shaped holes and two sets of anti-vibration bars (AVBs). There is a flow distribution baffle plate (FDB) between the tubesheet and the first TSP. The U-bend region of the tubes installed in rows 1-8 was thermally treated after bending to reduce stress.

3.1.2 Operating Experience

The last two SG inspections at Surry, Unit No. 2 were in spring 2017 (End of Cycle 27 (EOC27)) and fall 2018 (EOC28). More information regarding the SG inspections is available in the spring 2017 and fall 2018 SG Tube Inspection Reports (Reference 2) and (Reference 3), respectively. The most recent eddy current inspections of SG B were in spring 2017.

The following existing degradation mechanisms have been detected in the Surry, Unit No. 2 SGs: AVB wear, TSP wear, FDB wear, foreign object (FO) wear, circumferential primary water stress corrosion cracking (PWSCC) within the tubesheet, and pitting.

On April 17, 2012, the U.S. Nuclear Regulatory Commission (NRC) approved permanent alternate repair criteria for cracking found near the tube ends of the Surry, Unit No. 2 SGs. Specifically, the H* amendment revised TS 6.4.Q to exclude portions of the SG tube more than 17.89 inches below the top of the SG tubesheet from the normal inspection and plugging criteria (Reference 4).

Circumferential PWSCC above the H* depth was first detected in the Surry, Unit No. 2 SGs in 2015 during the EOC26 inspections (Reference 5). These indications, within the hot-leg tubesheet, were detected in three tubes in SG C. All three tubes were plugged. In 2017, the EOC27 inspections of SGs A, B, and C did not detect any form of stress corrosion cracking (SCC) and all tubes met the tube integrity performance criteria. No tubes required plugging. In 2018, the EOC28 inspections of SGs A and C did not detect any form of SCC and all tubes met the tube integrity performance criteria. No tubes required plugging. Table 1 provides a summary of tubes plugged in each SG by degradation mechanism.

Table 1: Surry, Unit No. 2 SG Tube Degradation Mechanisms

Degradation Mode	SG-A	SG-B	SG-C	Total
AVB Wear	1	5	10	16
Foreign Object	11	7	20	38
Pitting	11	0	1	12
SCC (excluding tube-end)	0	0	3	3
Tube-end SCC	3	1	2	6
Other*	4	6	14	24

* Shop plugs, restricted tubes, permeability variations, and 11 tubes plugged due to incomplete hydraulic expansions required due to a One-Time Alternative Repair Criteria.

The three Surry, Unit No. 2 SGs were placed in service in 1980 and have a total of 99 tubes that have been plugged (30 in SG A, 19 in SG B, and 50 in SG C).

The degradation assessment for Surry, Unit No. 2 includes the following potential degradation mechanisms for Alloy 600TT SG tube material: axial outside diameter stress corrosion cracking (ODSCC) at TSPs, circumferential ODSCC at the top of the tubesheet (TTS), and axial PWSCC at the TTS. Therefore, the SG inspection strategy for Surry, Unit No. 2 includes inspections for these potential degradation mechanisms with specialized eddy current probes. The licensee considered axial ODSCC at TSPs bounding for axial ODSCC at other locations in the SGs (i.e., dents, dings, and TTS) and made bounding assumptions in their operational assessment (OA) (see Section 3.3.2 of this safety evaluation) to account for axial ODSCC at other locations in the SGs.

The Alloy 600TT fleet is known to have some tubes with potentially higher residual stress that are more susceptible to SCC. Between 2005 and 2008, a total of 16 tubes with potentially high residual stress were identified in the Surry, Unit No. 2 SGs. All 16 tubes are in high row tubes (i.e., rows 9 and higher) with two tubes in SG B and 14 tubes in SG C. The inspection strategy for the tubes remaining in service in the Surry, Unit No. 2 SGs includes full-length bobbin coil examinations and array probe examinations of areas more susceptible to degradation.

The secondary-side activities for the Surry, Unit No. 2 SGs in the EOC27 (2017) refueling outage included sludge lancing, FO search and retrieval (FOSAR) at the top of tubesheet, and visual examination of accessible locations having eddy current possible loose part (PLP) indications, in SGs A, B, and C. In addition, a visual examination of accessible steam drum components and structures, including the feedring, feedring j-nozzles, and the primary and secondary moisture separators was performed in SG C.

The secondary-side activities for the Surry, Unit No. 2 SGs in the EOC28 (2018) refueling outage included a visual examination of accessible locations having eddy current PLP indications at the TTS, in the annulus, and in the no-tube lane in SGs A and C. In addition, a visual examination of the upper steam drum components and structures, including the feedring exterior, the upper tube bundle, and the moisture separators components was performed in SG B. Inspection of the seventh TSP was completed via probe insertions through the primary moisture separators. There have been no known foreign material introductions into either the primary or secondary systems since the 2017 refueling.

The licensee stated in the proposed LAR that deposit loading has been aggressively managed since performing a deposit minimization treatment in 2009, and no adverse trends in deposit loading currently exist in any Surry, Unit No. 2 SG. Also, there have been no chemistry excursions in the operating interval since 2017 at Surry, Unit No. 2. In addition, based on continuous main condenser air ejector and Nitrogen-16 main steam radiation monitors, and supplemented by daily leakage calculations, the licensee reported that no primary-to-secondary leakage has been detected in any Surry, Unit No. 2 SG.

3.2. Proposed TS Changes

3.2.1 Current TS Requirements

Surry, Unit Nos. 1 and 2 share TSs, and the "Steam Generator (SG) Program" in TS 6.4.Q provides the SG tube inspection requirements for both Surry, Unit Nos. 1 and 2. TS 6.4.Q.4.b requires, in part, that, "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)."

TS 6.4.Q.4.b.1, b.2, and b.3 define the SG tube inspection requirements for the first, second, third, and subsequent inspection periods following SG installation. Specifically, TS 6.4.Q.4.b.3 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." Surry, Unit No. 2 is currently in the third inspection period.

3.2.2 Description of Proposed TS Changes

The LAR proposes to add a footnote to TS 6.4.Q.4.b stating that the SG inspections scheduled for Unit 2 SG B in the spring 2020 refueling outage may be deferred on a one-time basis to the fall 2021 refueling outage. Specifically, TS 6.4.Q.4.b would be annotated by an asterisk at the end of the first sentence in the paragraph. A footnote with an asterisk at the bottom of the page would state: "* As approved by Amendment Nos. 299 and 299, the inspection of Surry Unit 2 SG B may be deferred, on a one-time basis, from the Surry Unit 2 spring 2020 refueling outage (S2R29) to the Surry Unit 2 2021 fall refueling outage (S2R30)."

3.3 Staff Evaluation of Proposed TS Changes

The NRC staff evaluation of the proposed exigent 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. Therefore, this safety evaluation should not be considered precedent setting for future routine plant amendments or generic industry licensing actions related to SG inspection intervals.

The NRC staff evaluation of the proposed one-time TS changes focused on the potential for affecting SG tube integrity since maintaining SG tube integrity ensures the plant will meet its SG Program related TS, thereby protecting the public health and safety. In particular, the staff evaluation assessed whether the licensee's amendment request demonstrates that the structural integrity performance criterion (SIPC) and accident-induced leakage performance criterion (AILPC) will be met for Cycle 30, which ends in fall 2021. These criteria are defined in TS 6.4.Q.2.

The Surry, Unit No. 2 operating experience has shown tube degradation from circumferential PWSCC within the tubesheet, wear at various support structures, wear from FOs, and pitting. All these mechanisms were evaluated as existing mechanisms in the OA, although not all have been identified in SG B, and pitting was detected only in the 1990s. The OA included circumferential ODSCC at the TTS, axial PWSCC at the TTS, and axial ODSCC at TSPs as potential mechanisms. The first two of these mechanisms have occurred in Surry, Unit No. 1 SGs, and all three have been observed at other units with Alloy 600TT tubing. The OA was performed using a deterministic analysis for most mechanisms, and a probabilistic analysis for two potential mechanisms. Table 2 summarizes the degradation mechanisms included in the OA and the corresponding assessment methods.

Table 2: Summary of Operational Assessment Methods for Surry, Unit No. 2 EOC30

Existing Mechanisms		
Mechanism	**OA Method**	**Basis for Inclusion**
Circumferential PWSCC in the tubesheet	Deterministic Worst-case single-tube	Detected in three tubes in SG C EOC26
AVB Wear		Detected in previous inspections in SG B
Foreign Object Wear		Detected in previous inspections in SG B
TSP/FDB Wear		Detected in SG A and SG C
Pitting	Qualitative	Last detected in 1990s, in SG A and SG C
Potential Mechanisms		
Circumferential ODSCC at TTS	Deterministic Worst-case single-tube	Detected in one tube in Surry, Unit No. 1
Axial ODSCC at TSP	Full-bundle probabilistic	Detected at other plants with Alloy 600 TT
Axial PWSCC at TTS		Detected in one tube in Surry, Unit No. 1

The simplified OA method uses the worst-case single-tube analysis from the Electric Power Research Institute's (EPRI) Integrity Assessment Guidelines (Reference 6) to provide a conservative estimate of the projected EOC condition considering all uncertainties at 0.95 probability and 50 percent confidence. The applicable uncertainties are for burst relation, material strength, and nondestructive examination (NDE) flaw sizing. The single tube methods are referred to as "worst-case degraded tube" methods as the most severely flawed tube is selected for evaluation. The worst-case degraded tube OA methods involve determining the most limiting flaw at the beginning-of-cycle (BOC) and applying conservative flaw growth over the intended inspection interval to arrive at the end-of-cycle (EOC) flaw condition to determine if the SIPC and AILPC will be met at EOC30. The probabilistic analysis approach, also referred to as "full bundle analysis," uses probabilistic models with distributions of flaw size and growth to determine the probability of burst and leakage. The projected EOC results are compared with the SIPC and AILPC acceptance criteria.

3.3.1 Evaluation of Existing Tube Degradation Mechanisms

Circumferential PWSCC Within the Tubesheet

SCC was detected at Surry, Unit No. 2 for the first time in EOC26. Three indications of circumferential PWSCC were identified at tube expansion bulges within the tubesheet in SG C. No SCC has been detected to date in SG A or SG B, and no additional cracking indications have been identified in SG C since EOC26. In EOC27, all the tubes in SG B were inspected with an array probe from the H* distance below the hot-leg and cold-leg TTS to the lowermost tube support, either an FDB plate or a TSP, whichever was lower. These inspections were also performed in SG A and SG C in EOC28.

The licensee performed a deterministic OA for circumferential ODSCC in SG B using the worst-case degraded tube methodology. For this method, the worst-case degraded tube must meet the SIPC and AILPC criteria. For Surry, Unit No. 2, the SIPC is a burst pressure of at least 4470 pounds per square inch (psi), which is three times the normal operating pressure differential (3xNOPD). For this degradation mechanism, the licensee used the maximum through-wall crack depth that prevents "pop-through" of the crack by ductile tearing and subsequent leakage. The pop-through relationship is described in Reference 6. As described in later sections of this safety evaluation, other degradation methods used the leakage value of 470 gallons per day (gpd) that was assumed in the limiting accident analysis.

The licensee's analysis assumed that the largest flaw existing at the BOC28 in spring 2017 had a depth equal to the depth of the 95 percent probability of detection (POD) curve for the inspection technique, as described in the Surry, Unit No. 2 degradation assessment. The analysis assumed a flaw with circumferential extent more than double that of the EOC26 indication in SG C to calculate the percent degraded area (PDA). (The three indications in SG C are the only indications of SCC in the Surry, Unit No. 2 SGs.) For the flaw growth rate, the licensee identified the maximum growth rate determined from the Alloy 600TT plant with the greatest number of SCC indications at the TTS, within the Alloy 600TT fleet. This maximum rate was adjusted for the Surry, Unit No. 2 hot-leg temperature and the resulting rate was classified as the 95th percentile growth rate, from which an average rate was also calculated. The average growth rate was applied to the operating cycle from EOC27 to EOC30, to calculate the EOC30 PDA for SIPC consideration, while the 95th percentile growth rate was used for AILPC consideration.

The calculated EOC30 equivalent crack size was smaller than the SIPC limiting crack size, so the licensee concluded that the SIPC was demonstrated until EOC30. For leakage from pop-through, the licensee determined the allowable through-wall (TW) extent, based on the relationship between circumferential extent and pop through limits at main steam line break conditions as shown in Figure 3-1 of the OA (Attachment to Enclosure 1 of Reference 1). By comparing the calculated EOC30 worst-case maximum depth TW to the pop-through limit, the licensee projected no leakage before EOC30, and that the AILPC would, therefore, be met.

The NRC staff reviewed the licensee's assessment for circumferential PWSCC within the tubesheet of Surry, Unit No. 2 SG B. This is the only form of SCC detected at Surry, Unit No. 2, and it has not been detected in SG B. The staff finds that the licensee made some conservative assumptions in the OA analysis, such as assuming that the tubesheet had no strengthening effect on burst pressure or pop-through leakage, basing the BOC flaw size on the 95th percentile POD of the inspection technique, and assuming a much larger crack length than the longest PWSCC flaw detected at Surry, Unit No. 2. Other assumptions made in the OA analysis were reasonable, such as using a calculated average flaw growth rate based on industry operating experience from the Alloy 600TT plant with the bounding fleet operating experience. The licensee's applied methodology from industry guidelines concluded that the SIPC and AILPC would be met with margin. Therefore, the staff finds the licensee's assessment acceptable for circumferential PWSCC within the tubesheet.

Wear at AVBs

Wear at AVBs has occurred in all three of the Surry, Unit No. 2 SGs, and five tubes in SG B have been plugged due to AVB wear indications. In the most recent inspections of SG B (EOC27) and SGs A and C (EOC28), there were indications of wear in each SG and at all four AVBs, but no tubes were plugged as a result. In SG B, the deepest wear indication returned to service was 17 percent TW, as sized with a bobbin probe using an EPRI-qualified examination technique. The licensee's OA for AVB wear was performed using the worst-case single tube method, as discussed previously. Based on NDE sizing uncertainty, the licensee adjusted the largest indication returned to service in EOC27 (17 percent TW) to a value of 29.8 percent TW, for use as the BOC28 depth for the assessment. To project the maximum TW depth at EOC30, the licensee applied a growth rate of 2.0 percent TW/effective full power years (EFPY), which was based on the highest value among the three Surry, Unit No. 2 SGs and resulted in a maximum projected depth of 38.8 percent TW at EOC30. This maximum projected EOC depth met the structural limit of 64 percent TW for Surry, Unit No. 2. Therefore, the licensee concluded that the SIPC and AILPC would be met for AVB wear until the next inspection of SG B at EOC30.

Wear at Support Plates (TSP and FDB)

No tubes have been plugged in the Surry, Unit No. 2 SGs due to wear at TSPs or the FDB, and this mechanism has not been detected in SG B. The FDB plate in the Surry SGs only covers the central portion of the SG tube bundle, and not the entire diameter of the tube bundle. For the BOC28 depth, the licensee assumed a 19 percent TW flaw (the 95 percent POD value for bobbin probe examinations) was present in the SG. A growth rate of 5 percent TW/EFPY (twice the largest growth rate observed during Cycle 28) was assumed. The projected EOC30 depth of 41.5 percent TW was below the 56.6 percent TW structural limit; therefore, the licensee concluded that the SIPC and AILPC would be met for TSP/FDB wear until the next inspection of SG B at EOC30.

Evaluation Summary for Wear at AVBs, TSPs, and the FDB

The NRC staff finds the licensee's evaluation of tube wear at AVBs, TSPs, and the FDB to be acceptable. Wear at these locations in the Surry, Unit No. 2 SGs has been effectively managed for many cycles without challenging tube integrity. SG B has a small number of AVB wear indications and only five tubes plugged due to AVB wear, and it has no indications of TSP/FDB wear. Wear at support structures is readily detected with standard eddy current examination techniques and wear sizing errors are considered in the projection of existing flaws until EOC30. The NRC staff found the determination of BOC flaw depth and growth rates acceptable because they are based on industry guidelines and conservative assumptions, and in line with rates determined for other plants with more data. Projections of EOC30 TW depths meet the SIPC with margin. For flaws of this type, for pressure loading only, satisfying the SIPC demonstrates that the AILPC will also be satisfied since the limiting accident induced pressure differentials are much less than the 3xNOPD. Therefore, the NRC staff finds that both SIPC and AILPC will be satisfied.

Foreign Object Wear

In addition to wear at support structures, Surry, Unit No. 2 has also experienced tube wear from FOs. In SG B, seven tubes have been plugged due to FO wear, and 31 tubes combined in SG A and SG C. The inspections performed in SG B in EOC27 included sludge lancing, post-lancing FOSAR of the TTS annulus and no-tube lane, and visual investigation of accessible locations with eddy current signals potentially related to FOs. A visual inspection of the secondary side of SG B was performed most recently in EOC28 and provided assurance that no new FOs were left in service. Following the EOC27, the licensee evaluated nine FO indications returned to service at that time, concluding that the eddy current signals had not changed during the prior two cycles and that the FOs were not still present at those locations. It was, therefore, assumed that no growth would occur. The largest of these flaws was 30 percent TW and 0.24 inch in length and satisfied the SIPC limit of 72 percent TW determined in the degradation assessment at that time. To account for new FO wear prior to EOC30, the licensee qualitatively assessed their experience with prior three-cycle inspection intervals (between 1992 and 2006) and the size of objects currently in SG C (small wires about 0.012 inches in diameter and 0.5 inches in length). The licensee acknowledges that future FO projections are difficult to predict, but Surry, Unit No. 2 experience indicates that FOs are unlikely to challenge the performance criteria until EOC30.

The NRC staff finds the licensee's analysis of FO wear reasonable based on its use of the information known about past FO wear in the Surry, Unit No. 2 SGs and the existing loose parts. The staff also acknowledges that predicting future loose part generation is not possible since past fleet-wide operating experience has shown that new loose part generation, transport to the SG tube bundle, and interactions with the tubes cannot be reliably predicted. However, plants can reduce the probability of loose parts by maintaining robust foreign material exclusion programs and applying lessons learned from previous industry operating experience with loose parts. Plants in general, including Surry, Unit No. 2, have demonstrated the ability to conservatively manage loose parts once they are detected by eddy current examinations or by secondary-side FOSAR inspections. If unanticipated aggressive tube wear from new loose parts should occur in a Surry, Unit No. 2 SG, operating experience has shown that a primary-to-secondary leak will probably occur, rather than a loss of tube integrity. In the event of a primary-to-secondary leak, the staff will interact with the licensee in accordance with established procedures in NRC Inspection Manual Chapter (IMC) 0327, "Steam Generator Tube Primary-to-Secondary Leakage," dated January 1, 2019 (Reference 7), to confirm the licensee's conservative decision making.

Pitting

Pitting was identified in the 1990s in Surry, Unit No. 2 SG A (11 tubes) and SG C (1 tube), just above the expanded portion in the cold-leg tubesheet. The last of these 12 tubes was plugged in 1999. Improvements in the management of sludge and water chemistry that were implemented in the mid-1990s have appeared to eliminate the cause of pitting (Reference 8). Pitting is therefore considered a dormant mechanism, although, as an existing mechanism, it continues to be in the scope of SG inspection. Based on the long-time dormancy and the understanding that the conditions leading to the pitting were corrected, the OA concluded that pitting will have no effect on meeting the SIPC and AILPC until EOC30. The NRC staff finds the assessment of pitting acceptable based on corrective actions taken when the pitting occurred, the extended period of time since it was last observed, and the capability of inspections to detect it.

3.3.2 Evaluation of Potential Tube Degradation Mechanisms

In addition to existing tube degradation mechanisms, the licensee considered the following potential degradation mechanisms:

- Circumferential ODSCC at the TTS
- Axial ODSCC at TSPs
- Axial PWSCC at the TTS

The licensee provided the rationale for assessing these SCC mechanisms, none of which have been detected in the Surry, Unit No. 2 SGs. Circumferential ODSCC at the TTS was included because it was identified in one tube in a Surry, Unit No. 1 SG during a previous outage. Axial ODSCC at TSPs was included because it was considered to be bounding for axial SCC at other locations as discussed below in the evaluation of this mechanism. Axial PWSCC at the TTS was included because it was detected in one tube in a Surry, Unit No. 1 SG during a previous outage.

As noted in Section 3.1.2 of this safety evaluation, the NRC staff previously approved an H* amendment for Surry, Unit No. 2 (Reference 4), which concluded that potential tube degradation beyond the H* depth in the tubesheet does not affect tube integrity. Therefore, the licensee's evaluation of potential mechanisms and this safety evaluation do not consider potential tube degradation between the Surry, Unit No. 2 H* distance and the tube end.

Both PWSCC and ODSCC have occurred at various locations at different plants in the Alloy 600TT fleet. 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. SCC in SG tubes is also accelerated by higher residual stress in the tubes. The Alloy 600TT fleet is known to have some tubes with higher residual stress that are more susceptible to cracking. Two such tubes have been identified in Surry, Unit No. 2 SG B, none in SG A, and 14 in SG C. In ECO27, along with full-length bobbin probe inspection of all tubes, the high-stress tubes in SG B were examined with an array probe for the entire hot-leg straight length.

Although the licensee has not detected these three potential forms of SCC in Surry, Unit No. 2, periodic eddy current examinations with specialized probes are performed to detect such cracking. These specialized probes are used for examination of 100 percent of the hot-leg and cold-leg tubes in the tubesheet region and can detect circumferential ODSCC and axial PWSCC. The specialized probes are also used to examine the entire hot-leg straight section of the known high-stress tubes and can detect axial ODSCC. (The full-length bobbin probe examinations can detect axial ODSCC, but with a lower probability than the specialized probes.) The licensee considered the examination methods for each mechanism in the assessments described below.

Circumferential ODSCC at the TTS

The licensee assessed this mechanism using a deterministic worst-case single tube method. For the largest undetected crack size at BOC, the licensee used a value based on the array probe detection capability for circumferential ODSCC flaws during qualification of the examination technique. Since only one indication of this type has been detected at Surry, Unit No. 1, for the flaw growth rate, the licensee identified the maximum growth rate determined from the Alloy 600TT plant with the greatest number of SCC indications at the TTS, within the Alloy 600TT fleet. This maximum rate was adjusted for the Surry, Unit No. 2 hot-leg temperature and the resulting rate was classified as the 95th percentile growth rate, from which an average rate was also calculated. The average growth rate was applied to the operating cycle from EOC27 to EOC30, to calculate the EOC30 PDA for SIPC consideration, while the 95th percentile growth rate was used for AILPC consideration. The SIPC and AILPC were met in both cases, so the licensee concluded that leakage would not occur and that the SIPC would also be met until EOC30.

Axial ODSCC at TSPs

The licensee assessed this mechanism with a full-bundle probabilistic analysis following the methodology in Reference 6. The simulation generates Monte Carlo projections of detected and undetected flaws for multiple operating cycles, considering inspection POD, new flaw initiation, and flaw growth, to calculate burst and leakage probabilities. This analysis addresses potentially unknown high-stress tubes by using a POD from bobbin probe examination, which is used to inspect the TSP intersections of all tubes, rather than using the array probe POD, which has been used to inspect TSP intersections only in the known high-stress tubes. A site-specific

POD curve was developed using eddy current noise measurements from SG B according to industry guidelines in Reference 9. While the NRC staff has not reviewed Reference 9, the staff found the resulting POD and growth rate curves developed by the licensee to be reasonable and conservative with 95th percentile values of approximately 80 percent TW and 17 percent TW/EFPY, respectively. The model was benchmarked to assume detection of axial ODSCC in EOC27, despite no past detections. Since a site-specific crack growth rate does not exist, the simulation used the EPRI default upper bound crack growth rate distribution, adjusted for Surry, Unit No. 2 temperatures. The analysis assumed a bounding crack length of the full TSP thickness. The resulting distributions of EOC30 worst-case degraded tube burst pressures and accident-induced leakage met the SIPC and AILPC.

Axial PWSCC at the TTS

The licensee assessed this mechanism using the same full-bundle probabilistic analysis used for axial ODSCC at TSPs to project future burst probability and accident-induced leakage. The same site-specific methodology was applied, but in this case, the POD curve was developed based on the array probe, which was used to examine all the tubes at the TTS in the last two inspections (EOC25 and EOC27). The model was benchmarked in the same way as was done for axial ODSCC but used the EPRI typical default growth rate distribution. The crack length was set to a value considered to be bounding for this mechanism. The resulting distributions of EOC30 worst-case degraded tube burst pressures and accident-induced leakage met the SIPC and AILPC.

Evaluation Summary for Potential Mechanisms

The NRC staff reviewed the licensee’s deterministic and probabilistic evaluation of potential mechanisms. The staff considers the licensee’s choice of potential SCC mechanisms acceptable since it considers mechanisms detected at the other Alloy 600TT plant with the same SG design (i.e., Surry, Unit No. 1), and cracking mechanism detected at multiple plants in the Alloy 600TT fleet (i.e., axial ODSCC at support plates). The staff considers the deterministic evaluation of circumferential ODSCC at the TTS acceptable based on conservative assumptions in determining the BOC crack size and the use of the largest maximum depth growth rate from the plant experiencing the greatest number of SCC indications, as the basis for assumed growth rates at Surry, Unit No. 2. The staff considers the probabilistic evaluation assumptions to be conservative for Surry, Unit No. 2, since they were benchmarked to assume detection of cracks in the last inspection of SG B, bounding crack lengths were assumed, and an appropriate growth rate was applied. The calculated probability of burst for each potential mechanism considered satisfies the SIPC margin requirements until the EOC30. The AILPC is also satisfied until EOC30 because the analyses predicted no leakage.

Based on the evaluation discussed above in Sections 3.3.1 and 3.3.2, the NRC staff concludes that there is reasonable assurance that both the tube structural integrity and leakage integrity performance criteria will be met for all tubes with existing known degradation and potential degradation until the EOC30.

3.4 Cycle 30 Mitigating Strategies

Surry TS 4.13.B has a requirement to “Verify primary to secondary LEAKAGE is ≤ 150 gallons per day through any one SG.” The current primary-to-secondary leakage administrative limit at Surry, Unit No. 2 is 75 gpd for initiating actions to shut down within 24 hours. In Section 3.2, “Mitigation Strategy,” of the LAR Enclosure 1, the licensee states that during operating Cycle 30, the administrative limit will be reduced from 75 gpd to 50 gpd attributable to Surry, Unit No. 2 SG B. The licensee also describes how leakage is measured and the actions that are already taken in response to increasing primary-to-secondary leakage. The NRC staff finds the lower primary-to-secondary leakage administrative limit to be acceptable because it provides for faster plant shutdown in the event of a primary-to-secondary leak.

3.5 10 CFR 50.36(c) Requirements

The licensee's application provides a summary of its operational assessment and discusses RCS operational leakage and the SG tube integrity requirements. The NRC staff confirmed that these requirements are not affected by the proposed change. Further, since a licensee's technical specifications are derived from the analyses and evaluation included in the licensee's UFSAR and amendments, the NRC staff finds that the licensee has evaluated, in an acceptable manner, that the proposed change does not affect the applicable UFSAR analyses. In addition, the licensee did not purpose any change to its current TS LCOs and SRs, and further stated that all applicable regulatory requirements will continue to be satisfied as a result of the proposed license amendments. Since the TS requirements remain unaffected by the proposed change, the modified TS 6.4.Q.4.b conforms to the 10 CFR 50.36 regulations and, as a result, operation of the facility in a safe manner is not affected and is, therefore, acceptable.

3.6 Technical Evaluation Conclusion

Based on the above, the NRC staff finds the proposed TS changes to be acceptable because the TS, as amended, will continue to meet 10 CFR 50.36(c). The regulations contained in 10 CFR 50.36 require that TS include items in specified categories, including LCOs, SRs, and administrative controls. The proposed changes to the licensee's TS administrative controls do not require any changes to the LCOs and SRs requirements. The Surry, Unit No. 2 TS continue to specify the LCOs and specify the remedial measures to be taken if one of these requirements is not satisfied. Further, the TS continue to specify the appropriate SRs to ensure the necessary quality of affected structures, systems, and components are maintained. Furthermore, the NRC staff finds that the licensee has demonstrated that there is reasonable assurance that the structural and leakage integrity of the Surry, Unit No. 2 SG B tubes will be maintained until the next SG tube inspections during Refueling Outage 30 in fall 2021. Therefore, the NRC staff concludes that the proposed changes to Surry TS 6.4.Q.4.b are acceptable.

4.0 EXIGENT CIRCUMSTANCES

The NRC's regulations contain provisions for the issuance of amendments when the usual 30-day prior public comment period cannot be met. These provisions are applicable under exigent circumstances. Consistent with the requirements 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; and (3) the NRC determines that the amendment involves no significant hazards consideration. As discussed in the licensee's application dated April 14, 2020, the licensee requested that the proposed amendments be processed by the NRC on an exigent basis.

Under the provisions in 10 CFR 50.91(a)(6), the NRC notifies the public in one of two ways when exigent circumstances exist: (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 22, 2020 (85 FR 22460).

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 social distancing recommendations provided by the United States Centers for Disease Control and Prevention in that they require workers to be in constant proximity to each other in a hot and radiological environment that exponentially increases the likelihood of individuals contracting COVID-19 and potentially inducing a rapid spread. The licensee further stated that on January 31, 2020, the U.S. Department of Health and Human Services declared a public health emergency for the United States to aid the nation's healthcare community in responding to COVID-19. Subsequently, the COVID-19 outbreak was characterized as a pandemic by the World Health Organization on March 11, 2020. On March 12, 2020, the Commonwealth of Virginia declared a state of emergency, and on March 13, 2020, President Donald Trump declared the COVID-19 pandemic a national emergency. Because the Surry TSs require operability of the SGs prior to entering Mode 4, the NRC staff determined that the need for these amendments does not allow for the standard public comment period but does allow for a 14-day comment period.

Based on the above, the NRC staff finds that the licensee made a timely application for the proposed amendments following its identification of the issue. In addition, the NRC staff finds that the licensee could not avoid the exigency due to the unforeseen circumstances related to the 2020 COVID-19 virus pandemic. Based on these findings and the determination that the amendments involve no significant hazards consideration as discussed below, the NRC staff has determined that a valid need exists for issuance of the license amendments 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.

As required by 10 CFR 50.91(a), in its application dated April 14, 2020, the licensee 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 change adds a note to TS 6.4.Q.4.b to permit a one-time deferral of the Surry Unit 2 SG B inspection from the Surry Unit 2 spring 2020 refueling outage (RFO) (S2R29) to the Surry Unit 2 fall 2021 refueling outage (S2R30). An operational assessment has been performed that concludes Surry Unit 2 SG B will continue to meet its specific structural and leakage integrity performance criteria throughout the operating period preceding the next inspection in fall 2021. In addition, the proposed change does not implement plant physical changes to any plant structure, system or component; hence, no new failure modes are introduced. Therefore, the probability of an accident previously evaluated is not significantly increased. Also, there is no significant increase in the consequences of an accident because the TS primary-to-secondary leakage limit is not being changed, and the SG tubes continue to meet the SG Program performance criteria and remain bounded by the plant's accident analyses.

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 change adds a note to TS 6.4.Q.4.b to permit a one-time deferral of the Surry Unit 2 SG B inspection from the Surry Unit 2 spring 2020 refueling outage (RFO) (S2R29) to the Surry Unit 2 fall 2021 refueling outage (S2R30). The proposed change does not alter the design function or operation of the SGs or the ability of a SG to perform its design function. The SG tubes continue to meet the SG Program performance criteria. No plant physical changes are being implemented that would result in plant operation in a configuration outside the plant safety analyses or design basis. The proposed change does not introduce any changes or mechanisms that create the possibility of a new or different kind of accident. Furthermore, Surry Unit 2 SG B will continue to meet its specific structural and leakage integrity performance criteria throughout the operating period preceding the next inspection in fall 2021. Finally, no new effects on existing equipment are created nor are any new malfunctions introduced.

   Therefore, based on the above evaluation, 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 change adds a note to TS 6.4.Q.4.b to permit a one-time deferral of the Surry Unit 2 SG B inspection from the Surry Unit 2 spring 2020 refueling outage (RFO) (S2R29) to the Surry Unit 2 fall 2021 refueling outage (S2R30). Extending the Surry Unit 2 SG B inspection schedule does not involve changes to any limit on accident consequences specified in the Surry licensing bases or applicable regulations, does not modify how accidents are mitigated, and does not involve a change in a methodology.

   A forward-focused operational assessment (OA) of Surry Unit 2 SG B was performed that demonstrates there is reasonable assurance the structural integrity and accident induced leakage performance criteria will remain satisfied in SG B throughout the period preceding the fall 2021 RFO inspection for a total operating duration of three cycles between primary side inspections. The OA also identified projected margin to the structural integrity and accident induced leakage performance criteria prior to the fall 2021 RFO for each evaluated degradation mechanism.

Therefore, operation of the facility in accordance with the proposed change will not involve a significant reduction in a margin of safety.

The NRC staff reviewed the licensee's no significant hazards consideration analysis. Based on this review and on the NRC staff's safety evaluation of the underlying license amendment request as discussed above, the NRC staff concludes that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff makes 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.

6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the NRC staff notified an official of the Virginia Division of Radiological Health of the proposed issuance of the amendments. On April 22, 2020, the State official confirmed that the Commonwealth of Virginia had no comments.

7.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission previously issued a proposed finding that the amendments involve no significant hazards consideration, published in the *Federal Register* on April 22, 2020 (85 FR 22460), and the agency has received no public comments on this finding. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Under 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

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

9.0 REFERENCES

1 Virginia Electric and Power Company, letter to U.S. Nuclear Regulatory Commission (NRC), "Virginia Electric and Power Comapny, Surry Power Station Units 1 and 2, Proposed License Amendment Request, One-Time Deferral of Surry Unit 2 Steam Generator "B" Inspection," April 14, 2020 (ADAMS Accession No. ML20105A223).

2 Virginia Electric and Power Company, letter to NRC, "Virginia Electric and Power Company, Surry Power Station Unit 2, Steam Generator Tube Inspection Report for the Spring 2017 Refueling Outage," October 26, 2017 (ADAMS Accession No. ML17310A235).

3 Virginia Electric and Power Company, letter to NRC, "Virginia Electric and Power Company, Surry Power Station Unit 2, Steam Generator Tube Inspection Report for the Fall 2018 Refueling Outage," May 16, 2019 (ADAMS Accession No. ML19154A353).

4 NRC, letter to Virginia Electric and Power Company, "Surry Power Station, Units Nos. 1 and 2, Issuance of Amendments Regarding Virginia Electric and Power Company License Amendment Request for Permanent Alternate Repair Criteria for Steam Generator Tube Inspection and Repair," April 17, 2012 (ADAMS Accession No. ML12109A270).

5 Virginia Electric and Power Company, letter to NRC, "Virginia Electric and Power Company, Surry Power Station Unit 2, Steam Generator Tube Inspection Report for the Fall 2015 Refueling Outage," May 10, 2016 (ADAMS Accession No. ML16141A117).

6 Electric Power Research Institute (EPRI), letter to NRC, "Request for Withholding of the following Proprietary Documents: Steam Generator Management Program: Steam Generator Integrity Assessment Guidelines, Revision 4, Product ID 3002007571," July 18, 2016 (ADAMS Accession No. ML16208A272).

7 NRC Inspection Manual, Inspection Manual Chapter 0327, "Steam Generator Tube Primary-to-Secondary Leakage," November 1, 2018 (ADAMS Accession No. ML18093B067).

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

9 EPRI, Model Assisted Probability of Detection Using R (MAPOD-R), Version 2.1, September 2017.

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SUBJECT: SURRY POWER STATION, UNIT NOS. 1 AND 2, ISSUANCE OF EXIGENT AMENDMENTS NOS. 299 AND 299 TO REVISE TECHNICAL SPECIFICATION 6.4.Q, "STEAM GENERATOR (SG) PROGRAM," TO ALLOW A ONE-TIME DEFERRAL OF THE SURRY UNIT NO. 2 SG "B" SPRING 2020 REFUELING OUTAGE INSPECTION (EPID NO. L-2019-LLA-0071) DATED MAY 7, 2020

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