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

August 10, 2020

Mr. James Barstow
Vice President, Nuclear Regulatory Affairs
and Support Services
Tennessee Valley Authority
1101 Market Street, LP 4A-C
Chattanooga, TN 37402-2801

SUBJECT: WATTS BAR NUCLEAR PLANT, UNIT 2 - ISSUANCE OF AMENDMENT NO. 40 REGARDING TECHNICAL SPECIFICATIONS FOR STEAM GENERATOR TUBE REPAIR SLEEVE (EPID L-2019-LLA-0209)

Dear Mr. Barstow:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 40 to Facility Operating License No. NPF-96 for the Watts Bar Nuclear Plant, Unit 2. This amendment is in response to your application dated September 30, 2019, as supplemented by letters dated November 21, 2019, and April 6 and June 12, 2020.

This amendment revises Watts Bar Nuclear Plant, Unit 2, Technical Specification (TS) 3.4.17, “Steam Generator (SG) Tube Integrity,” TS 5.7.2.12, “Steam Generator (SG) Program,” and TS 5.9.9, “Steam Generator Tube Inspection Report,” to allow the use of Westinghouse leak-limiting non-nickel banded Alloy 800 sleeves to repair degraded steam generator tubes as an alternative to plugging the tubes.

A copy of our related safety evaluation is also enclosed. A Notice of issuance will be included in the Commission’s biweekly *Federal Register* notice.

Sincerely,

/RA/

Michael J. Wentzel, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-391

Enclosures:
1. Amendment No. 40 to NPF-96
2. Safety Evaluation

cc: Listserv

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-391

WATTS BAR NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 40
License No. NPF-96

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

   A. The application for amendment by Tennessee Valley Authority (TVA, the licensee) dated September 30, 2019; as supplemented by letters dated November 21, 2019; and April 6 and June 12, 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 Facility Operating License No. NPF-96 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A as revised through Amendment No. 40 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. TVA 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 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Undine S. Shoop

Digitally signed by Undine S. Shoop
Date: 2020.08.10 13:38:31 -04'00'

Undine Shoop, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Facility Operating License
and Technical Specifications

Date of Issuance: August 10, 2020

ATTACHMENT TO AMENDMENT NO.40

WATTS BAR NUCLEAR PLANT, UNIT 2

FACILITY OPERATING LICENSE NO. NPF-96

DOCKET NO. 50-391

Replace page 3 of Facility Operating License No. NPF-96 with the attached revised page 3. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

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

Remove	Insert
3.4-38	3.4-38
3.4-39	3.4-39
5.0-15	5.0-15
5.0-16	5.0-16
5.0-17	5.0-17
--	5.0-17b
5.0-35	5.0-35

C. The 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

TVA is authorized to operate the facility at reactor core power levels not in excess of 3411 megawatts thermal.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A as revised through Amendment No. 40 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. TVA shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) TVA shall implement permanent modifications to prevent overtopping of the embankments of the Fort Loudon Dam due to the Probable Maximum Flood by June 30, 2018.

(4) PAD4TCD may be used to establish core operating limits until the WBN Unit 2 steam generators are replaced with steam generators equivalent to the existing steam generators at WBN Unit 1.

(5) By December 31, 2019, the licensee shall report to the NRC that the actions to resolve the issues identified in Bulletin 2012-01, "Design Vulnerability in Electrical Power System," have been implemented.

(6) The licensee shall maintain in effect the provisions of the physical security plan, security personnel training and qualification plan, and safeguards contingency plan, and all amendments made pursuant to the authority of 10 CFR 50.90 and 50.54(p).

(7) TVA shall fully implement and maintain in effect all provisions of the Commission approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The TVA approved CSP was discussed in NUREG-0847, Supplement 28, as amended by changes approved in License Amendment No. 7.

(8) TVA shall implement and maintain in effect all provisions of the approved fire protection program as described in the Fire Protection Report for the facility, as described in NUREG-0847, Supplement 29, subject to the following provision:

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.17 Steam Generator (SG) Tube Integrity

LCO 3.4.17 SG tube integrity shall be maintained

AND

All SG tubes satisfying the tube plugging or repair criteria shall be plugged or repaired in accordance with the Steam Generator Program.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each SG tube.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more SG tubes satisfying the tube plugging or repair criteria and not plugged or repaired in accordance with the Steam Generator Program.	A.1 Verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection.	7 days
	AND	
	A.2 Plug or repair the affected tube(s) in accordance with the Steam Generator Program.	Prior to entering MODE 4 following the next refueling outage or SG tube inspection.
B. Required Action and associated Completion Time of Condition A not met. OR SG tube integrity not maintained.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.17.1	Verify steam generator tube integrity in accordance with the Steam Generator Program.	In accordance with the Steam Generator Program.
SR 3.4.17.2	Verify that each inspected SG tube that satisfies the tube plugging or repair criteria is plugged or repaired in accordance with the Steam Generator Program.	Prior to entering MODE 4 following a SG tube inspection.

5.7 Procedures, Programs, and Manuals (continued)

5.7.2.12 Steam Generator (SG) Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during a SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging or repair of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected, plugged, or repaired to confirm that the performance criteria are being met.

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

    1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cooldown), all anticipated transients included in the design specification and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

(continued)

5.7.2.12 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 an 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 for all degradation mechanisms is not to exceed 150 gpd for each unfaulted SG. Leakage for all degradation mechanisms, excluding that described in Specification 5.7.2.12.c.2, is not to exceed 1 gpm in the faulted SG. Leakage for degradation mechanisms described in Specification 5.7.2.12.c.2 is not to exceed 4 gpm for the faulted SG.

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

c. Provisions for SG tube plugging or repair criteria. Tubes found by inservice inspection to contain a flaw in a non-sleeved region with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged or repaired.

The following alternate tube plugging shall be applied as an alternative to the 40% depth based criteria:

1. Tubes with service-induced flaws located in the portion of the tube from the top of the tubesheet to 1.64 inches below the top of the tubesheet, or from the bottom of the roll transition to 1.64 inches below the bottom of the roll transition, whichever is lower, shall be plugged. Tubes with service-induced flaws located below this elevation do not require plugging.

2. The voltage based methodology, in accordance with Generic Letter (GL) 95-05, shall be applied at the tube to straight leg tube support plate interface as an alternative to the 40% depth based criteria of Specification 5.7.2.12.c: Tubes shall be plugged in accordance with GL 95-05 or repaired.

Tube Support Plate Plugging Limit is used for the disposition of an Alloy 600 steam generator tube for continued service that is experiencing predominantly axially oriented outside diameter stress corrosion cracking confined within the thickness of the tube support plates and flow distribution baffles (FDB). At tube support plate intersections and FDB,

(continued)

5.7 Procedures, Programs, and Manuals

5.7.2.12 Steam Generator (SG) Program (continued)

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 1.64 inches below the bottom of the roll transition or 1.64 inches below the top of the tubesheet, whichever is lower at the tube inlet, to 1.64 inches below the bottom of the roll transition or 1.64 inches below the top of the tubesheet, whichever is lower at the tube outlet, and that may satisfy the applicable tube plugging or repair criteria. 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. After the first refueling outage following SG installation, inspect each SG at least every 24 effective full power months or at least every refueling outage (whichever results in more frequent inspections). In addition, inspect 100% of the tubes at sequential periods of 60 effective full power months beginning after the first refueling outage inspection following SG installation. Each 60 effective full power month inspection period 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. 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 or repair criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated.

(continued)

5.7 Procedures, Programs, and Manuals

5.7.2.12 Steam Generator (SG) Program (continued)

f. Provisions for SG Tube Repair Methods:

Steam generator tube repair methods shall provide the means to reestablish the RCS pressure boundary integrity of SG tubes without removing the tube from service. For the purposes of these Specifications, tube plugging is not a repair. All acceptable tube repair methods are listed below.

1. Westinghouse leak-limiting Non-Nickel Banded Alloy 800 sleeves, WCAP-15918-P, Revision 4, “Steam Generator Tube Repair for Combustion Engineering and Westinghouse Designed Plants with ¾ Inch Inconel 600 Tubes Using Leak Limiting Alloy 800 Sleeves.” A Non-Nickel Banded Alloy 800 sleeve shall remain in service for no more than five fuel cycles of operation starting from the outage when the sleeve was installed.

(continued)

5.9 Reporting Requirements (continued)

5.9.7 DG Failures Report

If an individual diesel generator (DG) experiences four or more valid failures in the last 25 demands, these failures and any nonvalid failures experienced by that DG in that time period shall be reported within 30 days. Reports on DG failures shall include the information recommended in Regulatory Guide 1.9, Revision 3, Regulatory Position C.4, or existing Regulatory Guide 1.108 reporting requirement.

5.9.8 PAMS Report

When a Report is required by Condition B or F of LCO 3.3.3, “Post Accident Monitoring (PAM) Instrumentation,” a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.9.9 Steam Generator Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.7.2.12, Steam Generator (SG) Program. The report shall include:

a. The scope of inspections performed on each SG,

b. Degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

d. Location, orientation (if linear), and measured sizes (if available) of service induced indications,

e. Number of tubes plugged or repaired during the inspection outage for each degradation mechanism,

f. The number and percentage of tubes plugged or repaired to date, and the effective plugging percentage in each SG,

g. The results of condition monitoring, including the results of tube pulls and in-situ testing.

h. Repair method utilized and the number of tubes repaired by each repair method.

(continued)

UNITED STATES
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WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 40 TO FACILITY OPERATING LICENSE NO. NPF-96

TENNESSEE VALLEY AUTHORITY

WATTS BAR NUCLEAR PLANT, UNIT 2

DOCKET NO. 50-391

1.0 INTRODUCTION

By letter dated September 30, 2019 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19274C001), as supplemented by letters dated November 21, 2019, and April 6 and June 12, 2020 (ADAMS Accession Nos. ML19326C088, ML20098D779, and ML20164A237, respectively); the Tennessee Valley Authority (TVA, the licensee), submitted a request for changes to the Watts Bar Nuclear Plant (WBN), Unit 2 (Watts Bar 2) Technical Specifications (TSs). The requested changes would revise Watts Bar 2, TS 3.4.17, "Steam Generator (SG) Tube Integrity," TS 5.7.2.12, "Steam Generator (SG) Program," and TS 5.9.9, "Steam Generator Tube Inspection Report," to allow the use of Westinghouse leak-limiting non-nickel banded Alloy 800 sleeves to repair degraded steam generator tubes as an alternative to plugging the tubes.

The supplements dated November 21, 2019; and April 6 and June 12, 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, the Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on January 7, 2020 (85 FR 734).

2.0 REGULATORY EVALUATION

2.1 System Description

In its letter dated September 30, 2019, Section 3.1, "System Description," the licensee described the SGs installed at Watts Bar 2, as follows:

> [Watts Bar 2] contains four Westinghouse Model D3 recirculating pre-heater type SGs. Each SG contains 4674 mill annealed (MA) Alloy 600 tubes that have an outer diameter of 0.75 inches with a 0.043-inch nominal wall thickness. These SGs are the same design as the original [Watts Bar 1] SGs. The [Watts Bar 1] SGs were replaced during the WBN U1R7 outage.

The [Watts Bar 2] SGs have a vertical shell and U-tube evaporator with integral moisture separating equipment. The reactor coolant flows through the inverted U-tubes, entering and leaving through the nozzles located in the hemispherical bottom head of the SG. The head is divided into inlet and outlet chambers by a vertical partition plate extending from the head to the tubesheet. Steam is generated on the shell side and flows upward through the moisture separators to the outlet nozzle at the top of the vessel. Details of the Unit 2 SGs are described in the WBN dual-unit updated final safety analysis report (UFSAR) Section 5.5.2.2 and UFSAR Figure 5.5-3 [ADAMS Accession No. ML19176A139].

The [Watts Bar 2] SGs contain a flow distribution baffle (FDB) plate located approximately eight inches above the top of the tube sheet. The tube holes located in the FDB design include an increased nominal tube-to-plate diametrical gap ranging from approximately 0.115 inches to 0.150 inches, compared to 0.023 inches nominal gap at the tube support plates (TSPs).

Materials of construction for the [Watts Bar 2] SG are provided in UFSAR Table 5.2-8. Materials are selected and fabricated in accordance with the requirements of the American Society of Mechanical Engineers (ASME) [Boiler and Pressure Vessel Code (Code)], Section III. The heat transfer tubes and the divider plate are inconel and the interior surfaces of the reactor coolant channel heads and nozzles are clad with austenitic stainless steel. The primary side of the tubesheet is weld clad with Inconel. The tubes are roller expanded for the full depth of the tubesheet after the ends are seal welded to the tubesheet cladding.

Tube and tubesheet stress analyses of the SG, which are discussed in UFSAR Section 5.2, confirm that the SG tubesheet will withstand the loading caused by loss of reactor coolant.

2.2 Description of Proposed Changes

The licensee proposed to revise TSs 3.4.17, 5.7.2.12, and 5.9.9, as follows (deleted text shown in strikethrough, additional text shown underlined):

Revise TS 3.4.17, Limiting Condition for Operation (LCO) 3.4.17, Condition A

LCO 3.4.17 SG tube integrity shall be maintained

AND

All SG tubes satisfying the tube plugging <u>or repair</u> criteria shall be plugged <u>or repaired</u> in accordance with the Steam Generator Program.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more SG tubes satisfying the tube plugging or repair criteria and not plugged or repaired in accordance with the Steam Generator Program.	A.1 Verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection. AND A.2 Plug or repair the affected tube(s) in accordance with the Steam Generator Program.	7 days Prior to entering MODE 4 following the next refueling outage or SG tube inspection

Revise TS 3.4.17, Surveillance Requirement (SR) 3.4.17.2

SURVEILLANCE		FREQUENCY
SR 3.4.17.2	Verify that each inspected SG tube that satisfies the tube plugging or repair criteria is plugged or repaired in accordance with the Steam Generator Program.	Prior to entering MODE 4 following a SG tube inspection.

Revise TS 5.7.2.12.a

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the “as found” condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The “as found” condition refers to the condition of the tubing during a SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging or repair of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected, ~~or~~ plugged, or repaired to confirm that the performance criteria are being met.

Revise TS 5.7.2.12.c

c. Provisions for SG tube plugging or repair criteria. Tubes found by inservice inspection to contain a flaw~~s~~ in a non-sleeved region with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged or repaired.

The following alternate tube plugging shall be applied as an alternative to the 40% depth-based criteria:

1. Tubes with service-induced flaws located in the portion of the tube from the top of the tubesheet to 1.64 inches below the top of the tubesheet, or from the bottom of the roll transition to 1.64 inches below the bottom of the roll,

transition, whichever is lower, shall be plugged. Tubes with service-induced flaws located below this elevation do not require plugging.

2. The voltage based methodology, in accordance with Generic Letter (GL) 95-05, shall be applied at the tube to straight leg tube support plate interface as an alternative to the 40% depth-based criteria of Specification 5.7.2.12.c: Tubes shall be plugged in accordance with GL 95-05 <u>or repaired</u>.

<u>Revise TS 5.7.2.12.d</u>

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 1.64 inches below the bottom of the roll transition or 1.64 inches below the top of the tubesheet, whichever is lower at the tube inlet, to 1.64 inches below the bottom of the roll transition or 1.64 inches below the top of the tubesheet, whichever is lower at the tube outlet, and that may satisfy the applicable tube plugging <u>or repair</u> criteria. 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. After the first refueling outage following SG installation, inspect each SG at least every 24 effective full power months or at least every refueling outage (whichever results in more frequent inspections). In addition, inspect 100% of the tubes at sequential periods of 60 effective full power months beginning after the first refueling outage inspection following SG installation. Each 60 effective full power month inspection period 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. 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 <u>or repair</u> criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated.

Add TS 5.7.2.12.f

f. Provisions for SG Tube Repair Methods:

Steam generator tube repair methods shall provide the means to reestablish the RCS pressure boundary integrity of SG tubes without removing the tube from service. For the purposes of these Specifications, tube plugging is not a repair. All acceptable tube repair methods are listed below.

1. Westinghouse leak-limiting Non-Nickel Banded Alloy 800 sleeves, WCAP-15918-P, Revision 3, "Steam Generator Tube Repair for Combustion Engineering and Westinghouse Designed Plants with ¾ Inch Inconel 600 Tubes Using Leak Limiting Alloy 800 Sleeves." A Non-Nickel Banded Alloy 800 sleeve shall remain in service for no more than five fuel cycles of operation starting from the outage when the sleeve was installed.

Revise TS 5.9.9

5.9.9 Steam Generator Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.7.2.12, Steam Generator (SG) Program. The report shall include:

a. The scope of inspections performed on each SG,

b. Degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

d. Location, orientation (if linear), and measured sizes (if available) of service induced indications,

e. Number of tubes plugged <u>or repaired</u> during the inspection outage for each degradation mechanism,

f. The number and percentage of tubes plugged <u>or repaired</u> to date, and the effective plugging percentage in each SG,

g. The results of condition monitoring, including the results of tube pulls and in-situ testing.

<u>h. Repair method utilized and the number of tubes repaired by each repair method.</u>

2.3 Applicable Regulatory Requirements and Guidance

The NRC staff reviewed the licensee's application to determine whether (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 activities proposed will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or the health and safety of the public. The

NRC staff considered the following regulatory requirements, guidance, and licensing and design-basis information during its review of the proposed changes.

Paragraph 50.36(c) of Title 10 of the *Code of Federal Regulations* (10 CFR) requires, in part, that TSs include items in the following categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) LCOs; (3) SRs; (4) design features; and (5) administrative controls.

Paragraph 50.36(c)(2)(ii) of 10 CFR requires, in part, that a TS LCO must be established for each item meeting one or more of the following criteria:

1. Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

2. A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3. A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

Paragraph 50.36(c)(3) of 10 CFR states, in part, that 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 LCO will be met.

Paragraph 50.36(c)(5) of 10 CFR states, in part, that administrative controls are provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner.

Appendix A, "General Design Criteria (GDC) For Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," establishes the minimum requirements for the principal design criteria for water-cooled nuclear power plants. The following GDC are applicable to this review:

- GDC 14, "Reactor coolant pressure boundary," which requires that the reactor coolant pressure boundary 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," which requires that the reactor coolant system and associated auxiliary, control, and protection systems be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

- GDC 30, “Quality of reactor coolant pressure boundary,” which requires that components which are part of the reactor coolant pressure boundary be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

- GDC 32, “Inspection of reactor coolant pressure boundary,” which requires that components which are part of the reactor coolant pressure boundary be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel.

Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,” to 10 CFR Part 50 requires, in part, a quality assurance program for the design, fabrication, construction, and operation of structures, systems, and components in nuclear plants. The pertinent requirements of Appendix B apply to all activities affecting the safety-related functions of those structures, systems, and components, including designing, purchasing, fabricating, handling, shipping, storing, cleaning, installing, inspecting, testing, operating, maintaining, repairing, refueling, and modifying.

Part 50.55a of 10 CFR requires, in part, that reactor coolant pressure boundary components meet the requirements for Class 1 components in Section III of the ASME Code. Part 50.55a further requires, in part, that throughout the service life of a pressurized water-cooled nuclear power facility, ASME Code Class 1 components meet the requirements, except design and access provisions and preservice examination requirements, set forth in Section XI, “Rules for Inservice Inspection of Nuclear Power Plant Components,” of the ASME Code, to the extent practical. Therefore, the SG sleeve repair method must be qualified in accordance with Section XI of the ASME Code, which refers to Section III of the ASME Code, as part of the design basis for the SG tubing. The sleeves must satisfy all applicable ASME Code, Section III limits for design, operating conditions, and accident loading conditions.

Regulatory Guide (RG) 1.121, “Bases for Plugging Degraded PWR [Pressurized-Water Reactor] Steam Generator Tubes” (ADAMS Accession No. ML003739366), provides guidance for determining the minimum SG tube wall thickness and for determining the repair criteria for SG tubes with sleeves. In accordance with RG 1.121, the margin of safety against tube rupture under normal operating conditions should not be less than three at any tube location where flaws have been detected. The margin of safety against tube failure under postulated accidents, such as a loss-of-coolant accident, main steam line break, or feed water line break concurrent with the safe shutdown earthquake, should be consistent with the margin of safety determined by the stress limits specified in Section III of the ASME Code.

3.0 TECHNICAL EVALUATION

The NRC staff evaluated the licensee’s application to determine whether the proposed changes are consistent with the regulatory requirements, guidance, and licensing and design-basis information discussed in Section 2.3 of this safety evaluation.

3.1 Evaluation of the Proposed Use of Leak-Limiting Sleeves

A sleeve is a tube segment that is inserted into an existing SG tube and expanded at both ends of the sleeve to form a structural joint. The leak-limiting sleeve is not designed to be leak tight.

Two leak-limiting non-nickel banded Alloy 800 sleeve designs were proposed for use in repairing SG tubes: a transition zone sleeve and a tube support sleeve. The transition zone sleeve is designed for tube degradation near the top of the tubesheet. The upper end of the transition zone sleeve forms a sleeve-to-tube joint by six equally spaced hydraulic expansions. The lower end of the transition zone sleeve forms a sleeve-to-tube joint by roll expansion of the sleeve into the tube within the tubesheet and includes a thermally sprayed nickel alloy band on the outside diameter surface of the sleeve. The thermally sprayed nickel alloy band contains a rough surface finish which enhances the strength of the rolled mechanical joint. The tube support sleeve is designed for tube degradation at TSP intersections or in the freespan region of SG tubes. The length of the transition zone and tube support sleeves are sized according to the length of the degraded tubing regions into which they are inserted.

3.1.1 Sleeve Installation

The licensee stated that the leak-limiting Alloy 800 sleeves will be installed in accordance with the processes provided by the vendor and described in the associated reports, which address sleeve design, qualification, installation methods, non-destructive examination, and as low as reasonably achievable radiation dose considerations. Installation of the sleeves will conform to ASME Code Section XI, IWA-4720, “Sleeving” (2007 Edition). The NRC staff notes that WCAP-15918-P, Revision 4 references ASME Code Section XI, IWA 4420, “Heat Exchanger Tube Sleeving” (1995 Edition). This difference reflects a change in paragraph numbering, but no change in technical content, for the later Edition of the ASME Code of record for Watts Bar 2 and is, therefore, acceptable to the NRC staff.

The sleeve is mounted on an expansion device and inserted into a tube for expansion. The expansion device is controlled and monitored to ensure consistent diametrical expansion. A hydraulic expansion tool is used at both ends of the tube support sleeves and at the top end of the transition zone sleeve, while a roll expander is used at the bottom end of the transition zone sleeve. The sleeve-to-tube joint formed by the hydraulic expansion generates the required structural and leakage integrity, while limiting the residual stresses in the parent tube. The torque of the roll expander is also monitored and controlled during installation. After the installation, all sleeve-to-tube joints undergo an initial acceptance and baseline inspection using the +POINT™ eddy current coil technology rotating probe; in addition, other eddy current methods will be considered for any complementary inspection capability.

3.1.2 Sleeve Materials Selection

The sleeve material, Alloy 800, is a nickel-iron-chromium alloy that was selected for its favorable properties, including corrosion resistance in both the primary and secondary side water chemistries. The Alloy 800 material is procured in accordance with the requirements of ASME Code, Section II, Part B, SB-163, NiFeCr Alloy, Unified Numbering System N08800, and Section III, Subsection NB-2000, “Material.” Additional restrictions will be applied to the alloying elements, the final annealing temperature, and yield strength for the Alloy 800 sleeve. This material is acceptable to the NRC staff because it is allowed by the ASME Code, which the NRC staff has approved.

3.1.3 Sleeve Qualification Testing

The licensee has performed qualification tests on samples with the sleeve-tube configuration in accordance with Appendix B to 10 CFR Part 50. The testing program included mechanical load tests, leakage tests, and corrosion tests. The mechanical load tests included leakage, axial load, load cycling, burst, and collapse. The tests were performed on sleeve/tube mock-ups that were constructed to the same dimensions as the installed sleeves in the field.

Mechanical Testing

The NRC staff notes that the sleeves that the licensee is proposing to use do not have a nickel band. As stated by the licensee, the nickel band has no effect on the structural capability of the sleeves as the rolled joint contact pressure and the presence of a “microlok” band contribute to this capability. Westinghouse performed testing utilizing sleeves with microlok bands and sleeves with microlok and nickel bands.

Axial load tests were performed to determine the structural integrity of the sleeve/tube joint. Axial loads are imposed as a result of the different thermal expansion rates of the leak-limiting Alloy 800 sleeve as compared to the Alloy 600 tube, and due to the differential pressure across the walls of the sleeve and tube. The test loads included the full range of loadings expected under normal power, transient, and accident conditions. The axial load tests showed that the leak-limiting Alloy 800 sleeve experiences only minor displacement, even when the parent tube is severed, and will not result in tube-to-tube contact in the U-bend area.

Collapse tests were also performed to show that the sleeve would not collapse following a loss of coolant accident. The collapse tests showed that the sleeve would not collapse even at secondary to primary differential pressures well above those experienced following a loss of coolant accident. The tests showed that once the pressure got high enough in the gap between the tube and the sleeve, the pressure would vent through the joint.

Load-cycling tests were performed to show that the structural and leakage integrity of the sleeve/tube joint will be maintained under cyclical differential thermal expansion and internal pressure in normal operating and transient conditions. The load-cycling tests included fatigue, thermal cycling, and mechanical load cycling. The load applied in the cycling tests was greater than three times the maximum operating differential pressure load. These tests showed that under various temperatures, the sleeve/tube joint is not significantly degraded by cyclic loads. The cycling tests confirm that slip during the initial heat-up is small, and the sleeve repositions itself inside of the parent tube to accommodate the thermal expansion without subsequent slip. As a part of the load cycling tests, the specimens were also tested for leakage integrity. The leak tests showed that the seal in the hydraulically expanded joints improved after load cycling.

Leak-rate tests were performed on the sleeve/tube assembly for various temperatures and pressures under normal operating and main steam line break conditions. The test results showed that the leakage from a single sleeve is extremely small relative to both the operational primary-to-secondary leakage limit in the plant technical specifications and the allowable leakage under accident conditions (see further discussion in Section 3.1.5 below).

The NRC staff finds the mechanical testing acceptable because it was performed under a quality assurance program and it verified that the load carrying capability of the sleeve/tube assembly met the regulatory acceptance criteria for structural and leakage integrity discussed above in Section 2.3 of this safety evaluation.

Corrosion Testing

Westinghouse performed various corrosion tests, including assessments of the leak-limiting Alloy 800 sleeves with full-length sleeved tube mock-ups. Sleeve/tube assemblies were pressurized with highly corrosive solutions and corrosion tests to assess the relative time to cracking of the sleeve/tube joint were also performed. The leak-limiting Alloy 800 sleeves did not develop any cracking in either the primary or secondary side tests and the leak-limiting Alloy 800 sleeve demonstrated higher corrosion resistance than the Alloy 600 parent tube.

The licensee stated that the leak-limiting Alloy 800 sleeves have not experienced service-induced degradation or leakage in nuclear power plants. The licensee also stated that besides leak-limiting Alloy 800 sleeves, Alloy 800 tubing has been used in PWR conditions in international nuclear plants with excellent results, based on the experience of over 200,000 tubes in service. The NRC staff notes that there have been limited occurrences of outer diameter stress corrosion cracking (ODSCC) in Alloy 800 tubing (not sleeves) in international plants.

The NRC staff also notes that the time for the initiation of corrosion in sleeve/tube assemblies was difficult to quantify accurately and that the accelerated lab testing method was unreliable for deterministic predictions. While the NRC staff did consider the corrosion tests to give a viable indicator of potential performance, at that time, the staff assumed a limited life expectancy for leak-limiting Alloy 800 sleeves.

Although the licensee submitted information that provides the NRC staff additional confidence in the corrosion resistance of the parent tube underneath the bottom roll joint, there is still some uncertainty in the long-term corrosion behavior of the sleeve and parent tube at the joints. Sleeves are inspected with a specialized eddy current probe each refueling outage to ensure that flaws in the sleeve/tube are detected and addressed. The NRC staff finds this acceptable because inspecting with a specialized eddy current probe each refueling outage will likely detect any degradation in a timely manner, should it occur.

Operating Experience

To address the issue of potential stress corrosion cracking (SCC) in the leak-limiting Alloy 800 sleeves, the licensee provided operating experience and laboratory data, as discussed below.

The licensee stated that Alloy 800 has not experienced primary or secondary side SCC. The NRC staff is not aware of any instances of SCC in Alloy 800 sleeves. However, as discussed above, there have been limited cases of ODSCC in Alloy 800 SG tubes in a few European PWRs. The ODSCC has occurred at the top of the tubesheet where a combination of higher stress and more aggressive local chemistry conditions can increase cracking susceptibility. Of the sleeve installations in Europe and Asia, most of them employ the PLUSS (PLUg replacing Sleeve which also Stabilizes) design, which is essentially identical to the subject Alloy 800 sleeve proposed for use at Watts Bar 2. To date, Westinghouse is not aware of any reports of parent tube degradation in the lower joint roll region for any Westinghouse sleeve design. For the Alloy 800 tubesheet sleeves in service over the past 20 years, as many as 8,750 of them were in service at one time with no reports of parent tube degradation. For the greater than 14,000 Alloy 800 tubesheet sleeve installations, the vast majority were installed at plants using mill-annealed Alloy 600 tubing.

The licensee compared the level of plastic deformation in the sleeve joint with that typically present at the top of the tubesheet and concluded that the sleeve joint would have a longer life than the original tube-to-tubesheet expansion zone. The licensee also evaluated Alloy 800 under model boiler conditions. In one out of the three boilers run with high chloride concentrations, modest pitting and shallow intergranular attack were observed. A fourth model run with sulfates found wastage but no SCC. Based on this data, the licensee concluded that Alloy 800 sleeves are sufficiently resistant to potential fault chemistries in the event that degradation in the parent tube leads to through-wall flaws.

The NRC staff finds that the operating experience and laboratory data submitted by the licensee provide additional assurance that the Alloy 800 sleeves and the parent tube behind the sleeve will demonstrate satisfactory corrosion performance. Sleeves are inspected with a specialized eddy current probe each refueling outage to ensure that flaws in the sleeve/tube are detected and addressed. The NRC staff finds this acceptable because inspecting with a specialized eddy current probe each refueling outage will likely detect any degradation in a timely manner, should it occur.

3.1.4 Sleeve Inspection

The licensee's sleeve examination program requires that no detectable degradation is present in the parent tube at the location of the hydraulic or roll expansions prior to sleeve installation. Sleeve installation will proceed only if the inspection finds these regions free from service-induced indications. This examination would ensure the area where the joints are to be established is free of detectable flaws, which provides additional assurance against degradation that could lead to leakage or compromise the integrity of the sleeve-to-tube joint.

To ensure that effective inspections of the sleeve/tube assembly could be performed, the capability to inspect these regions was assessed. The qualification program included fabricating samples with axially and circumferentially oriented notches representing flaws at each of the transitions and expansion zones. In addition, flaws in the pressure boundary portion of the sleeve and the parent tube away from the expansion regions were included in the sample set. The flaws included electro-discharge machined notches and a limited number of samples with cracking in the parent tube.

In previous industry submittals, the NRC staff has questioned the ability to inspect the region of the tube behind the nickel band of the nickel-banded sleeves. Consequently, other licensees have assessed the capability to inspect this region and the consequences of having undetected flaws in this region. For Watts Bar 2, the licensee has elected to submit a request to allow the use of Westinghouse leak-limiting non-nickel banded Alloy 800 sleeves. As stated previously, the nickel band is redundant because the rolled joint contact pressure and the presence of microlok contribute to the structural capability of the joint.

Based on the information provided in the license amendment request, the NRC staff finds the licensee's inspection program acceptable because the licensee (a) will be inspecting the parent tube at the location where the sleeve joints will be established to ensure that the region is free of detectable flaws prior to sleeving, (b) has demonstrated that severe degradation in the joints can be detected, (c) has determined that the axial load carrying capability of the joint is sufficient without a nickel band, and (d) has limited the amount of time that the sleeves will be in service to five refueling cycles.

3.1.5 Sleeve Structural Analysis

Westinghouse performed structural analyses in accordance with Appendix B to 10 CFR Part 50 and Section III of the ASME Code. The structural analyses included applied loads under normal and accident loading conditions. The analyses assumed two bounding tube configurations: (1) the tube is intact and (2) the tube is severed at the flaw location. In addition, the analyses assumed two bounding TSP configurations: (1) the tube is free to move past the TSPs and (2) the tube is locked in the first TSP and is prevented from axial motion. The structural analyses showed that stresses and fatigue factors in the worst sleeve/tube configuration satisfied the allowable stress and fatigue factor values in Section III of the ASME Code.

The structural analyses also included calculations for a minimum required sleeve thickness based on ASME Code, Section III. The calculations showed that the actual sleeve wall thickness is greater than the minimum required thickness and, therefore, acceptable. The percentage of sleeve wall thickness that could be acceptably degraded was also calculated. This calculation considered axial and circumferential cracking. The calculated amount of degradation that could be tolerated and still meet ASME limits was considered acceptable to the NRC staff because degradation of the sleeve is unlikely for the period of time the sleeve will be in service and the licensee will plug all flaws on detection.

The NRC staff finds that the licensee's sleeve structural analysis is consistent with the ASME Code and is, therefore, acceptable.

Under severe accident conditions in which primary system temperature may reach 1200 to 1500°Farenheit, the material properties of Alloy 800 are not significantly different from that of Alloy 600. As a result, the structural integrity of the leak-limiting Alloy 800 sleeve is commensurate with the integrity of the Alloy 600 parent tubing under severe accident conditions. This is acceptable to the NRC staff because the overall behavior of the sleeve/tube assembly will not change.

3.1.6 Sleeve Leakage Integrity

The sleeve-to-tube joint leakage was determined via laboratory testing to be small. For the leakage integrity assessment methodology, the licensee will conservatively assume all installed sleeves will leak under post-accident leakage conditions. The leak rate for each sleeve is an upper 95 percent confidence limit on the mean value of leakage for the appropriate temperature and pressure conditions. The licensee will combine the total sleeve leak rate with the total amount of leakage from all other sources (i.e., alternate repair criteria and non-alternate repair criteria indications) for comparison against the limit on accident-induced leakage, as specified in the UFSAR for all design basis accidents. The NRC staff finds that the licensee's leakage integrity assessment methodology is acceptable because it assumes a conservative estimate of leakage from each sleeve, combines this estimate with the leakage from all other sources, and then compares the combined value against the acceptance limits.

3.1.7 Conclusion

Based on the above, the NRC staff finds that the licensee has demonstrated the acceptability of the leak-limiting non-nickel banded Alloy 800 sleeve repair in accordance with Appendix B to 10 CFR Part 50, GDCs 14, 15, 30, and 32 of Appendix A to 10 CFR Part 50, RG 1.121, and the ASME Code. As noted above, the leak-limiting non-nickel banded Alloy 800 sleeves are limited to five refueling cycles of operation after installation.

3.2 Evaluation of Proposed Changes to the Technical Specifications

As discussed in Section 2.2 of this safety evaluation, the licensee proposed to revise TSs 3.4.17, 5.7.2.12, and 5.9.9 to allow for the option to repair degraded SG tubes and to define the acceptable method of repair. As discussed in Section 3.1 of this safety evaluation, the NRC staff reviewed the licensee's proposed use of leak-limiting non-nickel banded Alloy 800 sleeves to repair degraded SG tubes and found it to be an acceptable alternative to plugging those tubes, as currently required in the Watts Bar 2 TSs. As such, the NRC staff finds that the proposed changes to TSs 3.4.17, 5.7.2.12, and 5.9.9 satisfy the requirements of 10 CFR 50.36(c)(2), 10 CFR 50.36(c)(3), and 10 CFR 50.36(c)(5), and are, therefore, acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Tennessee State official was notified of the proposed issuance of the amendment on May 29, 2020. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes requirements with respect to installation or use of facility components located within the restricted area as defined in 10 CFR Part 20 and changes SRs. 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 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 January 7, 2020 (85 FR 734). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

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

Principal Contributor: A. Huynh

Date: August 10, 2020

SUBJECT: WATTS BAR NUCLEAR PLANT, UNIT 2 - ISSUANCE OF AMENDMENT NO. 40 REGARDING TECHNICAL SPECIFICATIONS FOR STEAM GENERATOR TUBE REPAIR SLEEVE (EPID L-2019-LLA-0209) DATED AUGUST 10, 2020

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