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

July 26, 2021

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 1 - ISSUANCE OF AMENDMENT NO. 147 REGARDING CHANGE TO STEAM GENERATOR TUBE INSPECTION FREQUENCY AND ADOPTION OF TECHNICAL SPECIFICATION TASK FORCE (TSTF) TRAVELER TSTF-510 (EPID L-2020-LLA-0161)

Dear Mr. Barstow:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 147 to Facility Operating License No. NPF-90 for the Watts Bar Nuclear Plant (Watts Bar), Unit 1. This amendment is in response to your application dated July 17, 2020, as supplemented by letters dated October 13, 2020, and March 30, 2021.

This amendment revises the Watts Bar, Unit 1, Technical Specification 5.7.2.12, "Steam Generator (SG) Program," and TS 5.9.9, "Steam Generator Tube Inspection Report," to reflect a change to the SG tube inspection frequency, and changes due to the adoption of Technical Specifications Task Force (TSTF) Technical Change Traveler TSTF-510, Revision 2, "Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection."

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

Sincerely,

/RA/

Kimberly J. Green, Senior Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-390

Enclosures:
1. Amendment No. 147 to NPF-90
2. Safety Evaluation

cc: Listserv

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-390

WATTS BAR NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 147
License No. NPF-90

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 July 17, 2020, as supplemented by letters dated October 13, 2020, and March 30, 2021, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-90 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. 147 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 amendment is effective as of the date of its issuance, and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

David J. Wrona

Digitally signed by David J. Wrona
Date: 2021.07.26 12:55:39 -04'00'

David J. Wrona, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Operating License

Date of Issuance: July 26, 2021

ATTACHMENT TO AMENDMENT NO. 147

WATTS BAR NUCLEAR PLANT, UNIT 1

FACILITY OPERATING LICENSE NO. NPF-90

DOCKET NO. 50-390

Replace page 3 of Facility Operating License No. NPF-90 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 revised pages. The revised pages are identified by amendment number and contain vertical marginal lines indicating the area of change.

Remove Pages	Insert Pages
3.4-43	3.4-43
3.4-44	3.4-44
5.0-15	5.0-15
5.0-16	5.0-16
--	5.0-16a
5.0-32	5.0-32

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

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

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

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

TVA is authorized to operate the facility at reactor core power levels not in excess of 3459 megawatts thermal.

(2) <u>Technical Specifications and Environmental Protection Plan</u>

The Technical Specifications contained in Appendix A as revised through Amendment No. 147 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) <u>Safety Parameter Display System (SPDS) (Section 18.2 of SER Supplements 5 and 15)</u>

Prior to startup following the first refueling outage, TVA shall accomplish the necessary activities, provide acceptable responses, and implement all proposed corrective actions related to having the Watts Bar Unit 1 SPDS operational.

(4) <u>Vehicle Bomb Control Program (Section 13.6.9 of SSER 20)</u>

During the period of the exemption granted in paragraph 2.D.(3) of this license, in implementing the power ascension phase of the approved initial test program, TVA shall not exceed 50% power until the requirements of 10 CFR 73.55(c)(7) and (8) are fully implemented. TVA shall submit a letter under oath or affirmation when the requirements of 73.55(c)(7) and (8) have been fully implemented.

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 criteria shall be plugged 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 criteria and not plugged 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 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
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 criteria is plugged 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 an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged, 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, 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.

   2. Accident induced leakage performance criterion: The primary-to-secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage for all degradation mechanisms is not to exceed 150 gpd for each unfaulted SG. Leakage for all degradation mechanisms is not to exceed 1 gpm in the faulted SG.

(continued)

5.7 Procedures, Programs, and Manuals

5.7.2.12 Steam Generator (SG) Program (continued)

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

c. Provisions for SG tube plugging criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube plugging criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. A degradation assessment shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

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

2. After the first refueling outage following SG installation, inspect each SG at least every 96 effective full power months. Tube inspections shall be performed using equivalent to or better than array probe technology. For regions where a tube inspection with array probe technology is not possible (such as due to dimensional constraints or tube specific conditions), the tube inspection techniques applied shall be capable of detecting all forms of existing and potential degradation in that region. In addition, the minimum number of tubes inspected at each scheduled inspection shall be the number of tubes in all SGs divided by the number of SG inspection outages scheduled in each inspection period as defined in a and b 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

(continued)

5.7 Procedures, Programs, and Manuals

5.7.2.12 Steam Generator (SG) Program (continued)

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.

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

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

3. If crack indications are found in any SG tube, then the next inspection for each affected and potentially affected SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever results in more frequent inspections). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary-to-secondary LEAKAGE.

(continued)

5.9 Reporting Requirements (continued)

5.9.7 EDG Failures Report

If an individual emergency diesel generator (EDG) experiences four or more valid failures in the last 25 demands, these failures and any nonvalid failures experienced by that EDG in that time period shall be reported within 30 days. Reports on EDG 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 during the inspection outage for each degradation mechanism,

f. The number and percentage of tubes plugged to date, and effective plugging percentage in each steam generator,

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

h. Discuss trending of tube degradation over the inspection interval and provide comparison of the prior operational assessment degradation projections to the as-found condition.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 147 TO FACILITY OPERATING LICENSE NO. NPF-90

TENNESSEE VALLEY AUTHORITY

WATTS BAR NUCLEAR PLANT, UNIT 1

DOCKET NO. 50-390

1.0 INTRODUCTION

By application dated July 17, 2020 (Reference 1), as supplemented by letters dated October 13, 2020 (Reference 2), and March 30, 2021 (Reference 3), the Tennessee Valley Authority (TVA, the licensee), submitted a license amendment request (LAR) to revise the Watts Bar Nuclear Plant (Watts Bar), Unit 1, technical specifications (TSs). The proposed changes would revise Watts Bar, Unit 1, TS 5.7.2.12, "Steam Generator (SG) Program," and TS 5.9.9, "Steam Generator Tube Inspection Report," to reflect a change to the SG tube inspection frequency from every 72 effective full power months (EFPM) to every 96 EFPM and to incorporate Technical Specifications Task Force (TSTF) Standard Technical Specifications (STS) Change Traveler TSTF-510, Revision 2, ''Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection" (References 4 and 5). The proposed changes related to TSTF-510 include a terminology change in TS 3.4.17, "Steam Generator (SG) Tube Integrity," that does not affect any requirements.

The licensee stated that it has reviewed TSTF-510, Revision 2, and the model safety evaluation dated October 19, 2011 (Reference 6) as announced in the Federal Register Notice dated October 27, 2011 (76 FR 66763), and has concluded that the justifications presented in TSTF-510 and the model safety evaluation are applicable to Watts Bar, Unit 1.

The supplements dated October 13, 2020, and March 30, 2021, 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 September 8, 2020 (85 FR 55508).

Enclosure 2

2.0 REGULATORY EVALUATION

2.1 Background

The tubes within an SG function as an integral part of the reactor coolant pressure boundary (RCPB) and, in addition, isolate fission products in the primary coolant from the secondary coolant and the environment. Steam generator tube integrity means the tubes are capable of performing this safety function in accordance with the plant design and licensing basis.

The fundamental regulatory requirements with respect to the integrity of the SG tubing are established in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities." Specifically, the general design criteria (GDC) in Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 provide regulatory requirements that state, in part, the RCPB shall "be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture" (GDC 14); "shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences" (GDC 15); shall be "designed, fabricated, erected, and tested to the highest quality standards practical" (GDC 30); shall be "designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized" (GDC 31); and shall 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" (GDC 32). These GDC are referenced in TSTF-510.

Watts Bar, Unit 1, was designed to meet the intent of the, "Proposed General Design Criteria for Nuclear Power Plant Construction Permits," published in July 1967. The Watts Bar construction permit was issued in January 1973. The Watts Bar Nuclear Plant Dual Unit Updated Final Safety Analysis Report (UFSAR), however, addresses the General Design Criteria (GDC) published as Appendix A to 10 CFR 50 in July 1971. Conformance with the GDCs is described in Section 3.1.2 of the UFSAR (Reference 7).

Section 50.55a, "Codes and standards," of 10 CFR, specifies that components that are part of the RCPB must meet the requirements for Class 1 components in Section III of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), except as provided in 10 CFR 50.55a(c)(2), (3), and (4). Section 50.55a further requires that throughout the service life of pressurized-water reactor (PWR) facilities like Watts Bar, Unit 1, ASME Code Class 1 components must meet the Section XI requirements of the ASME Code to the extent practical, except for design and access provisions, and pre-service examination requirements. This requirement includes the inspection and repair criteria of Section XI of the ASME Code. The Section XI requirements pertaining to in-service inspection of SG tubing are augmented by additional requirements in the TS.

2.2 Licensee's Requested Changes

The licensee proposed to revise TS Limiting Condition for Operation 3.4.17, CONDITION A, and Surveillance Requirement 3.4.17.2 to replace the word "repair" in the term "tube repair criteria" with the word "plugging."

The licensee proposed the following revisions to TS 5.7.2.12, which are identified in Enclosure 5 to Reference 3:

The introduction paragraph would be revised to delete the word, “provisions.”

TS 5.7.2.12.b.1 would be revised to add a closing parenthesis after the word, “cooldown”; delete the word “and” after the new end parenthesis; and delete the existing closing parenthesis after the word “specification.”

TS 5.7.2.12.b.2 would be partially revised from:

> For design basis accidents that have a faulted steam generator, accident induced leakage is not to exceed 1.0 gallon per minute (gpm) for the faulted steam generator and 150 gallons per day (gpd) for the non-faulted steam generators. For design basis accidents that do not have a faulted steam generator, accident induced leakage is not to exceed 150 gpd per steam generator.

to:

> Leakage for all degradation mechanisms is not to exceed 150 gpd for each unfaulted SG. Leakage for all degradation mechanisms is not to exceed 1 gpm in the faulted SG.

TS 5.7.2.12.c would be revised to replace the word “repair” in the term “tube repair criteria” with the word “plugging.”

TS 5.7.2.12.d would be revised to replace the word “repair” in the term “tube repair criteria” with the word “plugging.” Additionally, the last portion of the paragraph would be partially revised from:

> An assessment of degradation shall be performed […]

to:

> A degradation assessment shall be performed […]

TS 5.7.2.12.d.1 would be revised to change the word “replacement” to “installation.”

TS 5.7.2.12.d.2 would be replaced with the following, including new subparts a and b:

> After the first refueling outage following SG installation, inspect each SG at least every 96 effective full power months. Tube inspections shall be performed using equivalent to or better than array probe technology. For regions where a tube inspection with array probe technology is not possible (such as due to dimensional constraints or tube specific conditions), the tube inspection techniques applied shall be capable of detecting all forms of existing and potential degradation in that region. In addition, the minimum number of tubes inspected at each scheduled inspection shall be the number of tubes in all SGs divided by the number of SG inspection outages scheduled in each inspection period as defined

in a and b 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.

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

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

TS 5.7.2.12.d.3 would be revised to add “affected and potentially affected” to the first sentence after the words “for each,” and the parenthetical would be revised to replace the words “is less” and with the words “results in more frequent inspections.”

The licensee also proposed to revise TS 5.9.9 as follows:

Item b would be revised to remove the word “Active.”

Item e would be revised to remove the word “active.”

Item f would be revised as follows:

The number and percentage of tubes plugged to date, and effective plugging percentage in each steam generator,

Item h would be replaced with:

Discuss trending of tube degradation over the inspection interval and provide comparison of the prior operational assessment degradation projections to the as-found condition.

2.3 Regulatory Requirements and Guidance

The NRC staff considered the following regulations during its review of this LAR:

In 10 CFR 50.36, "Technical specifications," the NRC established its regulatory requirements related to the content of TSs. Pursuant to 10 CFR 50.36, TSs are required to include items in the following categories: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; and (5) administrative controls. The TSs for all current PWR licensees require that an SG Program be established and implemented to ensure that SG tube integrity is maintained. Programs established by the licensee, including the SG Program, are listed in the administrative controls section of the TS to operate the facility in a safe manner.

The NRC staff's guidance for the review of TSs is in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light Water Reactor] Edition" (SRP), Chapter 16.0, "Technical Specifications," Revision 3, dated March 2010 (Reference 8). As described therein, as part of the regulatory standardization effort, the NRC staff has prepared Standard Technical Specifications (STS) for each of the LWR nuclear designs. Accordingly, for Westinghouse Electric Company (Westinghouse) plant designs such as Watts Bar, Unit 1, the NRC staff's review includes consideration of whether the proposed changes are consistent with NUREG-1431, "Standard Technical Specifications, Westinghouse Plants" (Reference 9). NUREG-1431, Section 5.5, "Programs and Manuals," specifies programs that shall be established, implemented, and maintained, which the SG Program is one.

3.0 TECHNICAL EVALUATION

3.1 Background

For Watts Bar, Unit 1, the requirements for performing SG tube inspections and plugging are in TS 5.7.2.12, while the requirements for reporting the SG tube inspections and plugging are in TS 5.9.9. Steam generator tube integrity is maintained by meeting the performance criteria specified in TS 5.7.2.12.b for structural and leakage integrity, consistent with the plant design and licensing basis. TS 5.7.2.12.a requires that a condition monitoring assessment be performed during each outage in which the SG tubes are inspected, to confirm that the performance criteria are being met. TS 5.7.2.12.d includes provisions regarding the scope, frequency, and methods of SG tube inspections.

These provisions require that the inspections be performed with the objective of detecting flaws of any type that may be present along the length of a tube and that may satisfy the applicable tube repair criteria. The applicable tube repair criteria, specified in TS 5.7.2.12.c, 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. Watts Bar, Unit 1, TS 3.4.17 requires that SG tube integrity be maintained and that SG tubes be plugged if they satisfy the repair criteria in the Steam Generator Program. Note that this LAR proposes to change the term "repair criteria" to "plugging criteria" for consistency with current STS terminology.

TS 3.4.13, "RCS Operational LEAKAGE," includes a limit on operational primary-to-secondary leakage, beyond which the plant must be promptly shutdown. Should a flaw exceeding the tube plugging limit not be detected during the periodic tube surveillance required by the plant TS, the operational leakage limit provides added assurance of timely plant shutdown before tube structural and leakage integrity are impaired, consistent with the design and licensing bases.

As part of the plant's licensing basis, applicants for PWR licenses are required to analyze the consequences of postulated design-basis accidents, such as a SG tube rupture and a steam line break. These analyses consider primary-to-secondary leakage that may occur during these events and must show that the radiological consequences do not exceed the applicable limits of 10 CFR 50.67 or 10 CFR 100.11 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 Watts Bar, Unit 1, are being changed because of the proposed amendment and, thus, no radiological consequences of any accident analysis are being changed. The proposed changes maintain the accident analyses and consequences that the NRC has reviewed and approved for the postulated design-basis accidents for SG tubes.

3.1.1 Steam Generator Design

Watts Bar, Unit 1, has four Westinghouse Model 68AXP replacement SGs that were installed in 2006. Each SG contains 5,128 thermally treated Alloy 690 tubes (Alloy 690TT), which have a nominal outside diameter of 0.75 inches and a nominal wall thickness of 0.043 inches. The tubes were hydraulically expanded into the tubesheets, with the ends of each tube welded to the cladding on the primary side of the tubesheet. The straight portions of the tubes are supported by twelve advanced tube support grids (ATSGs). The U-bend portions of the tubes are supported by ventilated flat bar support trees with varying numbers of diagonal and vertical straps, depending on tube location. A thermal stress relief was applied to the full-length of the tubes in rows 1 through 38 following bending. The design includes an axial counter-flow preheater section.

3.1.2 Operating Experience

Since being placed in service in 2006, two degradation mechanisms have been detected in the Watts Bar, Unit 1, replacement SGs: tube wear from contact with the ATSGs and tube wear from contact with the U-bend supports. Tube wear from interaction with ATSGs is the only degradation mechanism that has caused tubes to be removed from service (i.e., plugged). Table 1 contains the plugging summary for the Watts Bar, Unit 1, SGs. Prior to operation, one tube was plugged due to overexpansion of the tube within the tubesheet, and one tube was plugged due to a volumetric indication attributed to tube installation. An operating experience review of degradation noted during inspections is provided below, based on References 10 through 13.

<u>Tube Inspection and Wear Degradation</u>

The first in-service inspection after SG installation was performed in spring 2008 during Refueling Outage 8 (1R8) and included a full-length bobbin probe inspection of all tubes. Rotating probe (+Point™) examinations were performed on special interest indications. The licensee plugged all six tubes with indications of ATSG wear, with the maximum depth measured at 13 percent through-wall (TW). No other degradation was detected.

Table 1: Watts Bar Unit 1 SG Tube Plugging

Reason for Plugging	SG1	SG2	SG3	SG4	Total
Pre-Service	0	1	0	1	2
Tubesheet overexpansion*	1	0	0	0	1
U-bend Support Structure Wear	0	0	0	0	0
ATSG Wear	2	4	7	7	20
Foreign Object (no wear detected)	0	0	0	6	6
Total	3	5	7	14	29
Percentage	0.06	0.10	0.14	0.27	0.14

*Plugged in 2008 (one cycle), with no change from pre-service inspection.

The licensee performed the second inservice SG inspection in fall 2012 during 1R11 and included approximately 58 percent of the in-service tubes with a bobbin probe. Rotating probe (+Point™) examinations were performed for confirmation and characterization of indications that could not be resolved by the bobbin probe. Array probes were used to supplement the bobbin probe in detection of foreign object (FO) wear at the top of the tubesheet (TTS) and detection of support structure wear up to the sixth cold-leg horizonal support. The inspection detected 72 indications of ATSG wear in 51 tubes, with the deepest indication measured at 32 percent TW. The licensee plugged the 14 tubes with wear indications of 15 percent TW and greater. The inspection also detected 8 indications of U-bend support wear in seven tubes, with a maximum measure depth of 12 percent TW. No tubes were plugged as a result of the U-bend support wear.

The third and most recent SG tube inspections were performed in spring 2017 during 1R14 and included 100 percent of the in-service tubes. Combination bobbin/array probes were used to inspect the full-length of all the tubes on the hot leg and cold leg except for the U-bend sections of rows 1 through 4. The U-bend sections of rows 1 through 4 were inspected only with bobbin probes because the radius of these bends is too small for passage of the array probes. The inspection detected a total of 419 ATSG wear indications and 59 U-bend support wear indications, with maximum measured depths of 37 percent TW for ATSG wear and 27 percent TW for U-bend support wear. No tubes were plugged during 1R14. Additional information about the three in-service inspections is available in the SG Tube Inspection Reports (References 10 through 13).

<u>Preventive Plugging</u>

In addition to the two tubes plugged prior to operation, seven tubes were preventively plugged and had no associated degradation. During 1R8, one tube was plugged due to a bulge from overexpansion of the tube into the tubesheet hole during fabrication. The bulge had been detected during the pre-service inspection and had not changed. During 1R11, six tubes were preventively plugged and stabilized to bound the location of an FO. No wear from the object was detected, but it could not be retrieved from the gap between the tube and adjacent stay rod.

<u>Corrosion Degradation</u>

Watts Bar, Unit 1, has not reported any indications of corrosion degradation, such as stress corrosion cracking (SCC), and to date, the NRC staff is unaware of any corrosion degradation in operating SGs with Alloy 690TT tubing. Regardless of the operating experience, the Watts Bar, Unit 1, TSs require that a degradation assessment be performed prior to each SG inspection to determine the type and location of flaws to which the tubes may be susceptible. Based on this assessment, the licensee determines which inspection methods need to be employed and at which locations. Therefore, the SG inspection strategy for Watts Bar, Unit 1, includes inspections with specialized eddy current probes for potential corrosion degradation mechanisms.

Primary Side Visual Inspections

Primary-side inspections for the Watts Bar, Unit 1, SGs during 1R11 and 1R14 included visual inspection of all previously installed tube plugs, the channel head bowl cladding, and the divider plate. The inspections identified no appearance of improper plug installation or degradation of cladding, divider plate, or divider plate welds.

Secondary Side Inspections

Secondary-side activities for the Watts Bar, Unit 1, SGs during 1R8 included TTS sludge lancing and foreign object search and retrieval (FOSAR) in the four SGs. One FO was identified in SG 1 and removed. In SG 4, approximately 65-70 objects were found and removed from the hot leg and cold leg TTS and in the feedwater flow distribution box. The objects were pieces of Flexitallic gasket material, weld slag, wire, and other objects that could not be identified. The source of the parts could not be determined. The feedwater distribution boxes have holes approximately 0.3 inches in diameter that limit the amount of foreign material that enters the tube bundle. The steam drum upper internals were visually inspected in all SGs with no degradation identified.

During 1R11, steam drum visual inspections were performed in SG 2 and SG 3 with no degradation identified. Visual inspection and FOSAR were performed at the TTS in the four SGs for detection of FOs and evaluating the effectiveness of sludge lancing. The inspection included an in-bundle visual examination of several columns in each SG. The inspections identified foreign material on the tubesheet in SG 4, with no wear identified. The three objects were pieces of wire, one of which was detected in the primary-side eddy current inspection. Two of the pieces were removed from the SG, but the third could not be removed, and six tubes bounding the location of the object were stabilized and plugged. Foreign objects were removed from the feedwater distribution boxes in each SG, with SG 4 having the largest number (i.e., 35).

During 1R14, secondary side FOSAR and visual inspections were performed at the TTS to detect FOs, assess hard deposit buildup, and to determine tubesheet cleaning effectiveness in all four SGs. The FOSAR inspections of all four SGs included visual examination of periphery tubes on the hot leg and cold leg annulus and center no-tube lane. Among the four SGs, a total of six FOs were removed from the tubesheet region and six objects remained. The remaining objects were small pieces of metal, wires, and bristles. These FOs were characterized and analyzed to demonstrate acceptability of continued operation without exceeding the tube integrity performance criteria.

3.2 Proposed TS Changes

3.2.1 Overview of Proposed Changes

The existing TS requirements related to SG tube integrity for Watts Bar, Unit 1, are described above in Sections 2.3 and 3.1. TS 5.7.2.12 contains the SG tube inspection requirements. TS

5.7.2.12.d.2 requires, in part, that "No SGs shall operate more than 72 effective full power months or three refueling outages (whichever is less) without being inspected." TS 5.7.2.12.d.2 also defines sequential periods during which 100 percent of the tubes must be inspected. Changing these inspection frequency requirements is the main reason for the amendment request. The proposed changes to the TS 5.7.2.12 also reflect the latest STS.

In addition to the SG Program in TS 5.7.2.12, TS 3.4.17 requires that tube integrity be maintained in accordance with the SG Program, and it includes a requirement to remove tubes from service by plugging if they meet the repair criteria. The licensee's proposed changes include replacing the term "repair criteria" with "plugging criteria" in accordance with current STS terminology.

Most of the proposed changes to the Watts Bar, Unit 1, TSs are related to the requirements for tube inspection frequency and adopting TSTF-510, Revision 2 (Reference 4 and 5). The remaining proposed changes are a plant-specific revision to the accident-induced leakage performance criterion (TS 5.7.2.12.b.2) and a new reporting criterion (TS 5.9.9.h) in the Steam Generator Tube Inspection Report (SGTIR).

The NRC has a process for revising the STS through changes proposed by the industry TSTF or NRC staff (Reference 14). This simplifies the review of changes to the STS and the adoption of the changes by licensees. The proposed changes are called "travelers" and can be "adopted" into plant-specific TSs through an LAR. With respect to the TS changes proposed for Watts Bar, Unit 1, Table 2 lists each TS with proposed revisions, and the reason a revision is proposed.

Table 2: Proposed Technical Specification Changes

Technical Specification	**Reason for Proposing Change**
3.4.17	TSTF-510
5.7.2.12	TSTF-510
5.7.2.12.b.1	TSTF-510
5.7.2.12.b.2	Plant-specific leakage criteria
5.7.2.12.c,	TSTF-510
5.7.2.12.d, 5.7.2.12.d.1, 5.7.2.12.d.2.a, 5.7.2.12.d.3	TSTF-510
5.7.2.12.d.2, 5.7.2.12.d.2.b	Change in tube inspection frequency
5.9.9.b, 5.9.9.e, 5.9.9.f	TSTF-510
5.9.9.h	Plant-specific reporting requirement

3.2.2 Changes Related to Tube Inspection Frequency

The LAR proposes to change Watts Bar, Unit 1 TS 5.7.2.12.d.2 of the SG Program from requiring SG inspections every 72 EFPM, or at least every third refueling outage (RFO) (whichever results in more frequent inspections), to every 96 EFPM. Each inspection would include 100 percent of the tubes. The change would remove the requirement to base the inspection frequency on the more restrictive metric between either the EFPM or RFO and use only the EFPM metric. Because Watts Bar, Unit 1, operates on an 18-month fuel cycle, this change would result in SG inspections being required at least every fifth RFO. The proposed change would delete the existing definition for first, second, third, and subsequent sequential inspection periods in TS 5.7.2.12.d.2. The proposed change would also eliminate the requirement to inspect 50 percent of the tubes by the RFO nearest the midpoint of the inspection period because each inspection would include 100 percent of the tubes.

Section 5.7.2.12.d.2.b is the new proposed requirement to perform an inspection of 100 percent of the tubes at least every 96 EFPM and defines this as the second and subsequent inspection periods. By using a common frequency of 96 EFPM for both the required SG inspections and the requirement to inspect 100 percent of the tubes, SG inspections will occur less frequently, but a complete inspection of all tubes in the SG will occur more frequently.

3.2.3 Changes Related to Adopting TSTF-510

The revision proposed to TS 5.7.2.12.d.2 would add new TS 5.7.2.12.d.2.a, which defines a first inspection period of 144 months following the first RFO after installation. This reflects the current definition of the first sequential period in the Watts Bar Unit 1 TS, as well as the definition in TSTF-510 for SGs with Alloy 690TT tubing. Because the first inspection period of 144 months has been completed for the replacement SGs at Watts Bar, Unit 1, this TS revision will not affect future inspections.

The following changes for consistency with TSTF-510 are proposed based on a change in wording from “repair criteria” to “plugging criteria,” additional minor wording and punctuation changes, and changes to the SG inspection reporting requirements:

- TS 3.4.17 – would change “repair criteria” to “plugging criteria” in three places
- TS 5.7.2.12 – would delete “provisions”
- TS 5.7.2.12.b.1 – would delete the word “and” and change the location of parentheses
- TS 5.7.2.12.c – would change “repair criteria” to “plugging criteria”
- TS 5.7.2.12.d – would change “repair criteria” to “plugging criteria” and “assessment of degradation” to “degradation assessment”
- TS 5.7.2.12.d.1 – would change “replacement” to “installation”
- TS 5.7.2.12.d.3 – would add “affected and potentially affected” and change “is less” to “results in more frequent inspections”
- TS 5.9.9 – would delete “active” from “active degradation” in items b and e of the Steam SGTIR, and would add the effective plugging percentage to the information reported in item f.

3.2.4 Other Proposed Changes

- TS 5.7.2.12.b.2 – would simplify the wording of the plant-specific accident induced leakage performance criterion but not change the leakage rate limits
- TS 5.9.9 – would add plant-specific item h to the SGTIR for discussing tube degradation trending over the inspection interval and comparing the prior operational assessment (OA) degradation projections to the as-found condition

3.3 <u>Staff Evaluation of TS Changes</u>

3.3.1 Evaluation of Changes Related to Tube Inspection Frequency

The licensee proposed changes to TS 5.7.2.12.d.2 and a new TS 5.7.2.12.d.2.b related to the tube inspection frequency. The NRC staff’s evaluation of the proposed TS changes focused on the potential for these changes to affect SG tube integrity, since maintaining SG tube integrity is a current TS requirement that plays a key role in protecting the public’s health and safety. In particular, the evaluation assessed whether the technical justification in the LAR, as supplemented, demonstrates that the structural integrity performance criterion (SIPC) and accident-induced leakage performance criterion (AILPC) will continue to be met with the revised inspection intervals proposed. These tube integrity criteria are defined in TS 5.7.2.12.b.

As noted above in Section 3.1.2, the only degradation mechanisms detected in the Watts Bar, Unit 1, SGs are tube wear from interaction with U-bend supports and ATSGs. Since being placed in service in 2006, only tube wear from ATSGs has resulted in tubes being plugged. Having wear from support structures as the only tube degradation mechanisms is consistent with industry operating experience, which has shown support structure wear to be the predominant tube degradation mechanism in SGs with Alloy 690TT tubing.

Revision 2 of the licensee's 1R14 OA, which was submitted with Reference 2, evaluates the U-bend support structure and ATSG wear mechanisms as existing mechanisms, and wear from FOs as a potential mechanism. To project the amount of future wear from U-bend supports and ATSGs, the OA includes both depth-based and a volume-based approach, and the methods include both deterministic (arithmetic) and statistical (Monte Carlo) methods. The original OA was performed at the time of the inspection with depth-based approaches only. The volume-based approach was added in Revision 2 to evaluate wear for the longer inspection periods proposed in the LAR.

The Monte Carlo analysis is a statistical method that uses random sampling from parameter distributions to determine the probability of burst (POB) and leakage at a future point in time. The Monte Carlo method uses a beginning-of-cycle flaw size, nondestructive examination (NDE) flaw size measurement uncertainties, flaw growth rate uncertainties, material property uncertainties, and burst equation relational uncertainties to determine whether the POB and leakage will meet the SIPC and AILPC of the plant TSs with a probability of 0.95 at a confidence level of 50 percent.

3.3.1.1 Evaluation of Existing Tube Degradation Mechanisms

<u>Wear at U-bend Support Structures</u>

Wear at U-bend support structures was first detected during the 1R11 inspection in 2012, with a total of eight indications on seven tubes in SG 2, SG 3, and SG 4, and a maximum depth of 12 percent TW. In the 1R14 inspection in 2017, U-bend support structure wear was detected in all four SGs, with a maximum wear depth of 27 percent TW. No tubes have been plugged in the Watts Bar, Unit 1, SGs as a result of U-bend support structure wear.

To project the maximum TW depth at 1R19, the licensee used a volumetric approach to model wear growth. The volumetric growth model projects the change in both volume and depth of the wear using measured wear depths and the mechanical process that causes the wear. A benchmarking process was performed for measured versus predicted wear depths to establish that depth predictions using the volumetric model were conservative relative to the measured values. Due to the low number of indications for U-bend support wear, a single growth-rate analysis was performed for all four SGs.

For an assumed flaw length of 2.5 inches, the licensee calculated a maximum projected wear depth of 60.6 percent through-wall at 1R19, which meets the structural limit of 62 percent TW. The structural limit is based on the limiting TS criterion of three times the normal operating pressure differential (3xNOPD) and includes material and burst pressure equation uncertainties. The licensee calculated a corresponding burst pressure of 3,807 pounds per square inch (psi) at a 95 percent probability and 50 percent confidence level, which meets the performance criterion of 3,798 psi corresponding to the structural limit. Since the maximum projected flaw depth at 1R19 met the SIPC, and because the pressure differential for accident-induced leakage is less than 3xNOPD, the licensee concluded both the SIPC and AILPC would be met for U-bend support wear until the next inspection at 1R19.

Wear at Advanced Tube Support Grids

Wear at ATSGs was first detected during the 1R8 inspection in all four SGs, with a total of nine indications on six tubes, and a maximum depth of 13 percent TW. In the 1R11 inspection there were 72 indications of ATSG wear in 51 tubes among the four SGs. The maximum wear depth in 1R11 was 32 percent TW. In the 1R14 inspection there were 419 indications of ATSG wear in all four SGs, with a maximum wear depth of 37 percent TW. In total, 20 tubes have been plugged due to ATSG wear.

To project the maximum TW depth at 1R19, the licensee used the same volumetric approach described in the preceding section for U-bend support wear. The licensee stated that the volumetric approach can assume either a flat wear profile or a tapered wear profile, and that a tapered profile results in a less conservative structural depth limit (more allowable wear). This is because tapered wear removes less volume than flat wear for a given depth, and the structural limit calculation depends, in part, on the volume of material removed. In addition, the licensee stated that ATSG wear at Watts Bar, Unit 1, is primarily tapered, but conservatively treated as flat. In its projection of ATSG wear to 1R19, the licensee treated two of the wear indications as tapered wear after using three-dimensional thickness plots from the array probe to confirm the flaws were tapered.

The licensee calculated a maximum projected wear depth of 59.9 percent through-wall at 1R19, which meets the structural limit for flat wear of 63 percent TW. The licensee calculated a corresponding burst pressure of 3,918 psi at a 95 percent probability and 50 percent confidence level, which meets the performance criterion of 3,798 psi corresponding to the structural limit. Since the maximum projected flaw depth at 1R19 met the SIPC, and because the pressure differential for accident induced leakage is less than 3xNOPD, the licensee concluded both the SIPC and AILPC would be met for U-bend support wear until the next inspection at 1R19.

Evaluation Summary for Wear at U-bend Supports and ATSGs

Wear at these locations in the SGs has been effectively managed since installation of the replacement SGs in 1R7 without challenging tube integrity. Twenty tubes have been plugged due to ATSG wear, and no tubes have been plugged due to U-bend support wear, which is the only other existing degradation mechanism. All the tubes have been inspected at least two times with techniques qualified to detect the existing and potential mechanisms. Wear at support structures is readily detected with standard eddy current examination techniques. As explained in Reference 1, the licensee's projections of wear at 1R19 made conservative assumptions about the length of the wear flaws, the average operating cycle time, and the pressure differential (i.e., 3xNOPD) used to determine the SIPC. The projection methodology includes NDE flaw size measurement uncertainties, flaw growth rate uncertainties, material property uncertainties, and burst equation relational uncertainties.

The licensee's analysis concluded that the projected ATSG and U-bend support wear at 1R19 meet the SIPC with margin. The NRC staff finds the licensee's approach acceptable because it was based on industry guidelines and conservative assumptions. The use of tapered wear for two ATSG flaws reduces conservatism, but the staff considers it reasonable because the licensee confirmed the flaws are tapered and assumed all other flaws are flat. 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 3xNOPD. Therefore, both SIPC and AILPC are satisfied. Based on the preceding discussion, the NRC staff finds the licensee's evaluation of tube wear at U-bend supports and ATSGs to be acceptable.

3.3.1.2 Evaluation of Potential Degradation Mechanisms

Foreign Object Wear

Watts Bar, Unit 1, has not identified any tube wear from FOs using bobbin and array probes for detection of FOs and wear from FOs. Six tubes were preventively plugged in SG 4 during 1R11 due to an eddy current indication of a possible FO, but no object was found. Objects found during the 1R14 secondary-side inspections include bristles, weld slag, and other small metal objects. As explained in Reference 1, the main feedwater flow into the Watts Bar, Unit 1, SGs is discharged from a feedwater box through small diameter drilled holes that serve as FO strainers, thereby limiting the potential for FOs to be introduced in the SGs. The licensee performed a wear analysis for the objects known to be present based on past FOSAR activities. Based on the analysis the licensee concluded the SGs can operate at least five fuel cycles (7.5 EFPY), which is the time until the next planned SG inspection, before the object with the greatest potential to cause tube wear degradation could potentially exceed the tube integrity performance criteria. The analysis results are conservatively based on an assumed flaw length much greater than the flaw lengths expected from any wear that the objects might cause.

Stress Corrosion Cracking

The OA performed following 1R14 did not include any SCC mechanisms as potential mechanisms. This is consistent with Reference 15, which requires OA projections to be performed only for existing degradation mechanisms. To date, no SCC or other corrosion mechanisms have been identified in Alloy 690TT SG tubing. Nonetheless, the licensee is required by its TSs to perform a degradation assessment prior to each SG inspection and to perform inspections with inspection methods that are capable of detecting flaws of any type that may be present along the length of the tube. This TS requirement ensures that each SG inspection will look for all types of degradation that may be present, whether they are existing or potential.

Although no form of SCC has been detected in the Watts Bar, Unit 1, SGs, the licensee is adding a requirement to the TS 5.7.2.12.d.2 to perform periodic eddy current examinations with probes that are equivalent to or better than array probe technology.

The NRC staff notes that the enhanced detection achieved by inspection with advanced probes could reasonably be expected to provide a mitigating factor to increased operational time between inspections, by providing a more accurate assessment of the current tube condition. The NRC staff believes that such inspections are an important element of an inspection program supporting an increased interval between inspections. Regardless of the specific tubing alloy in an SG, detection of existing loose parts is enhanced by using advanced probes, such as the combination bobbin and array coil probe or other equivalent (or better) probes.

Evaluation Summary for Potential Mechanisms

The NRC staff finds the licensee's analysis of FO wear reasonable based on the mapping and evaluation of FOs in the SGs, previous operating experience, and a SG design (with FO strainers in the feedwater box) that inherently limits introduction of FOs into the Watts Bar, Unit 1, SGs. No FO wear has been identified, and the evaluation of known parts for potential wear concluded structural and leakage integrity would be maintained until the next planned inspection at 1R19.

The staff also acknowledges that predicting future FO and 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 FOs and loose parts by maintaining robust foreign material exclusion programs and applying lessons learned from previous industry operating experience. Plants in general, including Watts Bar Unit 1, have demonstrated the ability to conservatively manage FOs and loose parts once they are detected by eddy current examinations or by secondary-side FOSAR inspections. If unanticipated aggressive tube wear from new FOs or loose parts should occur in a Watts Bar Unit 1 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 Inspection Manual Chapter (IMC) 0327, "Steam Generator Tube Primary-to-Secondary Leakage," dated January 1, 2019 (Reference 16), to confirm the licensee's conservative decision making. Based on the licensee's evaluation of known FOs, and the preceding discussion of managing FOs and loose parts, the staff finds the licensee's evaluation acceptable.

For the reasons discussed in the preceding section, the NRC staff finds it acceptable that the licensee did not address SCC as a potential mechanism in the OA. The licensee will continue to perform inspections capable of detecting SCC and other corrosion mechanisms and has proposed a new requirement to use enhanced probe technology in the TS.

3.3.1.3 Evaluation Summary for Changes Related to Tube Inspection Frequency

The NRC staff finds the licensee's proposed changes to TS 5.7.2.12.d.2 and proposed new TS 5.7.2.12.d.2.b related to tube inspection frequency acceptable. This is based on the staff's review of the licensee's evaluation of existing and potential degradation mechanisms, and the combination of existing and proposed inspection requirements. Specifically, the licensee's degradation projections concluded the structural and leakage performance criteria will be met at the next planned inspection with margin. In addition, the TS will require 100 percent of the tubes to be inspected in periods of no more than 96 EFPM, using enhanced inspection probe technology.

3.3.2 Evaluation of Changes Related to Adopting TSTF-510

TS 3.4.17 Steam Generator (SG) Tube Integrity

The term "repair criteria" would be changed to "plugging criteria" in three places. This change was made in TSTF-510 for consistency in the meaning of tube repair throughout the TS. The NRC staff finds this change acceptable because it is consistent with TSTF-510.

TS 5.7.2.12 Steam Generator (SG) Program

The word “provisions” would be deleted from the introductory paragraph. This change was made in TSTF-510 because “provisions” was found to be unnecessary in that location. The NRC staff finds this change acceptable because it is consistent with TSTF-510.

TS 5.7.2.12.b.1

The word “and” would be deleted, and the location of a parenthesis would be changed as an editorial correction. The NRC staff finds this change acceptable because it is an editorial change consistent with TSTF-510.

TS 5.7.2.12.c

The term “repair criteria” would be changed to “plugging criteria.” As discussed above for TS 3.4.17, the NRC staff finds this acceptable because it is consistent with TSTF-510.

TS 5.7.2.12.d

The term “repair criteria” would be changed to “plugging criteria,” and the term “An assessment of degradation” would be changed to “A degradation assessment.” These changes were made in TSTF-510 for consistent terminology throughout the TSs and other industry documents. The NRC staff finds the changes acceptable because they are consistent with TSTF-510.

TS 5.7.2.12.d.1

The term “SG replacement” would be changed to “SG installation.” This change was made in TSTF-510 to make the wording applicable to both existing plants and new plants. The NRC staff finds this change acceptable because it is consistent with TSTF-510.

TS 5.7.2.12.d.2.a

New TS 5.7.2.12.d.2 would be added to define a first inspection period of 144 months following the first RFO after installation. The NRC staff finds this change acceptable because it reflects the current definition of the first sequential period in the Watts Bar, Unit 1, TS, as well as the definition in TSTF-510 for SGs with Alloy 690TT tubing. The NRC staff also notes that because the first inspection period of 144 months has been completed for the replacement SGs at Watts Bar, Unit 1, this TS revision will not affect future inspections.

TS 5.7.2.12.d.3

The phrase “affected and potentially affected” would be added to clarify the SG inspection requirements when crack indications are found. In addition, the phrase, “whichever is less” would be changed to “whichever results in more frequent inspections,” to clarify the intent for the subsequent inspection after finding crack indications. The NRC staff finds these changes acceptable because they are consistent with TSTF-510.

TS 5.9.9 Steam Generator Tube Inspection Report

The inspection reporting requirements would be changed by adding the effective plugging percentage to the information reported in item f. In addition, the word “active” would be deleted from the term “active degradation” in items b and e. The NRC staff finds these changes acceptable because they are consistent with TSTF-510.

Evaluation Summary for Changes Related to Adopting TSTF-510

Based on its review of the TS changes proposed for adopting TSTF-510, the NRC staff finds the proposed changes acceptable because they are consistent with TSTF-510 and with the other proposed TS revisions.

3.3.3 Evaluation of Other Proposed Changes

TS 5.7.2.12.b.2

The wording of the AILPC would be simplified, but the criteria are unchanged. For unfaulted SGs, leakage is not to exceed 150 gpd, and for the faulted SG, leakage is not to exceed 1 gpm. The NRC staff finds this change acceptable because the leakage criteria remain consistent with the assumptions made in Chapter 15 of the UFSAR and the limits in the STS and TSTF-510.

TS 5.9.9

The inspection reporting requirements would be changed by adding a plant-specific item h for discussing tube degradation trending over the inspection interval and comparing the prior OA degradation projections to the as-found condition. The NRC staff finds this change acceptable because the SGTIR would provide additional information for assessing whether the degradation projections continue to be conservative with respect to meeting the tube integrity requirements with margin.

Evaluation Summary for Other Proposed Changes

For the reasons given in the preceding paragraphs, the NRC staff finds the proposed changes to the AILPC and the addition of SGTIR reporting requirement h acceptable.

3.4 Technical Evaluation Conclusion

Based on the information submitted, the NRC staff finds there is reasonable assurance that the structural and leakage integrity of the Watts Bar, Unit 1, SG tubes will be maintained after making the proposed changes to TSs 3.4.17, 5.7.2.12, and 5.9.9. Additionally, the staff finds that the Watts Bar, Unit 1, TSs, as revised, meet the requirements in 10 CFR 50.36. Therefore, the NRC staff finds the proposed changes 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 March 25, 2021. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20, “Standards for Protection Against Radiation.” The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has made a finding that the amendment involves no significant hazards consideration, and there has been no material public comment on that finding, published in the *Federal Register* on September 8, 2020 (85 FR 55508). 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.

7.0 REFERENCES

1. Barstow, J., Tennessee Valley Authority (TVA), letter to U.S. Nuclear Regulatory Commission (NRC), “Application to Revise Watts Bar Nuclear Plant (WBN), Unit 1 Technical Specifications for Steam Generator Tube Inspection Frequency and to Adopt TSTF-510, ‘Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection,’ (WBN-390-TS-20-012),” July 17, 2020 (ADAMS Accession No. ML20199M346).

2. Polickoski, J. T., TVA, letter to U.S. NRC, “Supplement to Application to Revise Watts Bar Nuclear Plant (WBN), Unit 1 Technical Specifications for Steam Generator Tube Inspection Frequency and to Adopt TSTF-510, ‘Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection,’ (WBN-390-TS-20-012) (EPID L-2020-LLA-0161),” October 13, 2020 (ADAMS Accession No. ML20287A569).

3. Polickoski, J. T., TVA, letter to U.S. NRC, “Response to Request for Additional Information Regarding Application to Revise Watts Bar Nuclear Plant (WBN), Unit 1 Technical Specifications for Steam Generator Tube Inspection Frequency and to Adopt TSTF-510, ‘Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection,’ (WBN-390-TS-20-012) (EPID L-2020-LLA-0161),” March 30, 2021 (ADAMS Accession No. ML21091A151).

4. Technical Specifications Task Force (TSTF) Traveler TSTF-510, Revision 2, “Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection,” (ADAMS Accession No. ML110610350), March 1, 2011.

5. Mendiola, A. J., U.S. NRC, letter to Technical Specifications Task Force, “U.S. Nuclear Regulatory Commission (NRC) Staff Response [sic] To TSTF Letter Dated March 28, 2012, Regarding Correction to TSTF-510, Revision 2, ‘Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection,’” June 17, 2013 (ADAMS Accession No. ML13120A541).

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Date: July 26, 2021

SUBJECT: WATTS BAR NUCLEAR PLANT, UNIT 1 - ISSUANCE OF AMENDMENT NO. 147 REGARDING CHANGE TO STEAM GENERATOR TUBE INSPECTION FREQUENCY AND ADOPTION OF TECHNICAL SPECIFICATION TASK FORCE (TSTF) TRAVELER TSTF-510 (EPID L-2020-LLA-0161) DATED JULY 26, 2021

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