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

January 25, 2022

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. 60 REGARDING REVISION OF TECHNICAL SPECIFICATION REQUIREMENTS SPECIFIC TO THE MODEL D3 STEAM GENERATORS THAT WILL NO LONGER APPLY FOLLOWING STEAM GENERATOR REPLACEMENT (EPID L-2021-LLA-0043)

Dear Mr. Barstow:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 60 to Facility Operating License No. NPF-96 for the Watts Bar Nuclear Plant, Unit 2. This amendment is in response to your application dated March 11, 2021, as supplemented by letters dated August 11, 2021, and January 5, 2022.

This amendment revises the Watts Bar Nuclear Plant, Unit 2, technical specifications (TSs) to delete several requirements for steam generator (SG) tube inspection and repair methodologies that will no longer apply following installation of the replacement SGs. The amendment also revises TS 5.7.2.12.d.2 to reflect the SG inspection interval criteria for Alloy 690 thermally treated tubing. Additionally, the amendment revises Watts Bar, Unit 2, Facility Operating License No. NPF-96, Paragraph 2.C.(4) to delete the reference to PAD4TCD, which will no longer apply following the installation of the replacement SGs.

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. 50-391

Enclosures:
1. Amendment No. 60 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. 60
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 March 11, 2021, as supplemented by letters dated August 11, 2021, and January 5, 2022, 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. 60 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. Condition 2.C.(4) of Facility Operating License No. NPF-96 is hereby amended to read as follows:

   (4) FULL SPECTRUM LOCA Methodology shall be implemented when the WBN Unit 2 steam generators are replaced with steam generators equivalent to the existing steam generators at WBN Unit 1.

4. This amendment is effective as of the date of its issuance, and shall be implemented prior to entering Mode 4 during restart following the U2R4 refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

David J. Wrona

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Date: 2022.01.25 09:53:41 -05'00'

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
and Technical Specifications

Date of Issuance: January 25, 2022

ATTACHMENT TO AMENDMENT NO. 60

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 revised pages. The revised pages are identified by amendment number and contain a marginal line indicating the area of change.

Remove Pages	Insert Pages
3.4-38	3.4-38
3.4-39	3.4-39
5.0-15	5.0-15
5.0-16	5.0-16
5.0-16a	--
5.0-16b	--
5.0-17	5.0-17
5.0-17a	5.0-17a
5.0-17b	--
5.0-35	5.0-35
5.0-36	5.0-36

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 3459 megawatts thermal.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A as revised through Amendment No. 60 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) FULL SPECTRUM LOCA Methodology shall be implemented when 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 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.	7 days
	AND		
	A.2	Plug 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.	6 hours
	AND		
	B.2	Be in MODE 5.	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 a 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, 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 Procedures, Programs, and Manuals

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 is not to exceed 1 gpm in 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 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.

(continued)

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 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 72 effective full power months or at least every third refueling outage (whichever results in more frequent inspections). In addition, the minimum number of tubes inspected at each scheduled inspection shall be the number of tubes in all SGs divided by the number of SG inspection outages scheduled in each inspection period as defined in a, b, c and d below. If a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at this location and that may satisfy the applicable tube plugging criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated. The fraction of locations to be inspected for this potential type of degradation at this location at the end of the inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the inspection period after the determination that a new form of degradation could potentially be occurring at this location divided by the total number of times the SG is scheduled to be inspected in the

(continued)

5.7 Procedures, Programs, and Manuals

5.7.2.12 Steam Generator (SG) Program (continued)

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

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

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

3. 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 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 during the inspection outage for each degradation mechanism,

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

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

(continued)

5.10 Record Retention

(removed from Technical Specifications)

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 60 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 March 11, 2021 (Reference 1), as supplemented by letters dated August 11, 2021 (Reference 2), and January 5, 2022 (Reference 3), the Tennessee Valley Authority (TVA, the licensee) submitted a license amendment request (LAR) to the U.S. Nuclear Regulatory Commission (NRC, the Commission) to revise the Watts Bar Nuclear Plant (Watts Bar or WBN), Unit 2, Operating License and Technical Specifications (TSs). The requested changes would revise 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 remove requirements related to the F* methodology, voltage-based alternate repair criteria (ARC), and SG tube sleeving that will no longer apply following installation of the replacement SGs (RSGs). Additionally, TS 5.7.2.12.d.2 would be revised to reflect the TS changes in Nuclear Regulatory Commission-approved Technical Specification Task Force (TSTF)-510, Revision 2, "Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection," for Alloy 690 thermally treated (Alloy 690TT) tubing (References 4 and 5). The Watts Bar, Unit 2, Facility Operating License No. NPF-96, Condition 2.C.(4) would also be revised to delete the reference to the PAD4TCD computer program, which will no longer apply following installation of the RSGs.

The supplements dated August 11, 2021, and January 5, 2022, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on July 13, 2021 (86 FR 36781).

2.0 REGULATORY EVALUATION

2.1 System Description

The licensee plans to replace the Watts Bar, Unit 2, SGs during the Cycle 4 refueling outage (U2R4) scheduled for spring 2022. The Unit 2 RSGs are Westinghouse Model 68AXP, which are the same model of RSGs that were installed at Watts Bar, Unit 1, in 2006. The RSGs are vertical shell and U-tube heat exchangers with integral moisture separating equipment. Secondary coolant enters the shell side through a feedwater distribution box into a preheater section of the tube bundle. The reactor coolant flows through the inverted U-tubes, generating steam on the shell side that flows upward through the moisture separators to the outlet nozzle at the top of the vessel. The SG tubes function as an integral part of the reactor coolant pressure boundary (RCPB) and isolate fission products in the primary coolant from the secondary coolant and the environment. Therefore, SG tube integrity is an important safety barrier. SG tube integrity means the tubes are capable of performing these safety functions in accordance with the plant design and licensing basis.

The tubing material in the Watts Bar, Unit 2, original SGs is mill annealed Alloy 600 (Alloy 600MA). The RSGs have Alloy 690TT tubing material that is more corrosion resistant. Each of the four RSGs contains 5,128 tubes with an outside diameter of 0.75 inches and a nominal wall thickness of 0.043 inches. The tubes are supported horizontally by stainless steel advanced tube support grids. An upper bundle support system with diagonal and vertical strip assemblies provides support to the U-bend region of all tubes. Each SG has an integral preheater section with a flow distribution plate.

For SG design, 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 to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50 provide regulatory requirements that state, in part, the RCPB shall have "an extremely low probability of abnormal leakage…and of gross rupture" (GDC 14), "shall be designed with sufficient margin" (GDCs 15 and 31), shall be of "the highest quality standards practical" (GDC 30), and shall be designed to permit "periodic inspection and testing…to assess…structural and leak tight integrity" (GDC 32).

Section 50.55a of 10 CFR specifies that components which 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 2, 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 TSs.

Watts Bar, Unit 2, was designed to meet the intent of the "Proposed General Design Criteria for Nuclear Power Plant Construction Permits" published in July 1967, and the Watts Bar construction permit was issued in January 1973. The Watts Bar dual-unit Updated Final Safety Analysis Report (UFSAR) addresses the NRC GDC published as Appendix A to 10 CFR 50 in July 1971, including Criterion 4 as amended October 27, 1987. The licensee takes no exceptions to GDC 14, 15, 30, 31, and 32, related to SG tube integrity, as documented in UFSAR Section 3.1, "Conformance with the NRC General Design Criteria" (Reference 6).

For SG replacement projects, the NRC staff perform inspections to address these regulatory requirements related to tube integrity. Inspection Procedure (IP) 50001, "Steam Generator Replacement Inspection," (Reference 7), directs the development of a site-specific inspection plan for steam generator replacement and directs inspectors to use IP 71111.17T, IP 71111.18, and IP 71111.21M (References 8 - 10) for completion of baseline inspections. Inspectors use the site-specific inspection plan to select and review the safety-related aspects associated with the major phases of the SG replacement project. The IP 50001 also directs the inspector to confirm that the licensee completed engineering evaluations and design changes associated with SG replacements in conformance with the requirements of the facility license, the applicable codes and standards, licensing commitments, and the regulations.

The NRC staff's review of the proposed changes to the TSs and License Condition 2.C.(4) are documented within this safety evaluation (SE). However, this SE does not document the review of the actual SG replacement activity. As permitted by 10 CFR 50.59, "Changes, tests and experiments," a licensee may make changes to its facility described in the UFSAR, make changes to procedures as described in the UFSAR, and conduct tests or experiments not described in the UFSAR, without obtaining a license amendment, provided that it does not meet any of eight criteria in 10 CFR 50.59(c)(2). As described above, the NRC inspectors will confirm that the licensee completed engineering evaluations and design changes associated with the SG replacement in conformance with the requirements of the facility license, the applicable codes and standards, licensing commitments, and the regulations. These inspections are outside the scope of this SE, but will be documented in the appropriate inspection reports.

2.2 Licensee's Requested Changes

TVA proposed the following changes to Watts Bar, Unit 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 remove requirements related to the ARC (i.e., F*, Generic Letter (GL) 95-05 voltage-based, and SG tube sleeve repair) that will no longer apply following installation of the replacement SGs. Additionally, TVA proposed changes to TS 5.7.2.12.d.2 to reflect changes for the RSGs with Alloy 690TT tubing, consistent with TSTF-510, Revision 2. The proposed changes are summarized or shown below (deleted text shown in strikethrough, additional text shown underlined).

TS 3.4.17, "Steam Generator (SG) Tube Integrity"

The phrases "or repair" and "or repaired" would be deleted from:

- Limiting Condition for Operation (LCO) 3.4.17
- Condition A
- Required Action A.2
- Surveillance Requirement (SR) 3.4.17.2

TS 5.7.2.12, "Steam Generator (SG) Program"

- TS 5.7.2.12.a, "Provisions for condition monitoring assessments," would be revised to delete the phrases "or repair" and "or repaired," and "or" would be relocated as follows:

  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.

- TS 5.7.2.12.b.2, "Accident induced leakage performance criterion," would be revised as follows:

> Leakage for all degradation mechanisms~~, excluding that described in Specification 5.7.2.12.d.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.~~

- TS 5.7.2.12.c would be revised as follows to delete references to repairs, repair sleeves, and alternate plugging criteria:

> c. Provisions for SG tube plugging ~~or repair~~ criteria. Tubes found by inservice inspection to contain ~~a~~ flaws ~~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:~~

- TS 5.7.2.12.c.1 would be revised to delete the ARC to allow flaws below a certain depth within the tubesheet region

- TS 5.7.2.12.c.2, associated items a through e, and Note 1, associated with the GL 95-05 (Reference 11) voltage-based ARC for axial outside diameter stress corrosion cracking at tube support plate intersections, would be deleted

- TS 5.7.2.12.d, "Provisions for SG tube inspections," would be revised as follows to require inspection of the entire tube length within the tubesheet region and state that the tube-to-tubesheet weld is not part of the tube:

> 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~~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 ~~or repair~~ 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.

- TS 5.7.2.12.d.2 specifies the scope and frequency of inspections based on the tube material. It would be revised extensively from the TSTF-510, Revision 2, specifications for Alloy 600MA to reflect the specifications for Alloy 690TT tubing material. These changes can be summarized as follows:

  - The inspection period starting after the first refueling outage would be increased from "at least every 24 effective full power months (EFPM) or at least every refueling outage," to "at least every 72 EFPM or at least every third refueling outage."
  - Four new items (a, b, c, d) would be added to define the length of inspection periods over which 100 percent of the tubes must be inspected. The four paragraphs define four different periods that decrease with increasing age of the SGs (144, 120, 96, and 72 EFPM).
  - The minimum number of tubes in all SGs that must be inspected at each scheduled inspection would be specified according to the number of inspections scheduled in each period.
  - Requirements would be added for the minimum inspections required for potential new degradation mechanisms that have not been inspected for previously.

- TS 5.7.2.12.d.4, which contains inspection requirements associated with the GL 95-05 voltage-based ARC, would be deleted.

- TS 5.7.2.12.f, "Provisions for SG Tube Repair Methods," which includes provision 5.7.2.12.f.1 for installation of repair sleeves, would be deleted.

Section 5.9.9, "Steam Generator Tube Inspection Report"

- TSs 5.9.9.e and 5.9.9.f would be revised to delete "or repaired"

- TS 5.9.9.f would also be revised to delete the word "the" before the words "effective plugging percentage in each SG."

- Item 5.9.9.h, for listing repair methods utilized and the number of tubes repaired by each method, would be deleted

- All reporting requirements associated with the GL 95-05 voltage-based ARC would be deleted.

TS Bases

The licensee submitted conforming changes to the TS Bases for information. The staff noticed the proposed changes to the bases for TS 3.4.17 delete references to tube repair and, therefore, are consistent for RSGs with Alloy 690TT tubing at plants using TSTF-510, Revision 2.

Operating License Condition 2.C.(4)

The licensee proposed to revise the Watts Bar, Unit 2, Facility Operating License No. NPF-96, Condition 2.C.(4) by deleting the first sentence regarding PAD4TCD because it will no longer apply following installation of the RSGs. The revised license condition will read as follows:

FULL SPECTRUM LOCA Methodology shall be implemented when the WBN Unit 2 steam generators are replaced with steam generators equivalent to the existing steam generators at WBN Unit 1.

2.3 Applicable Regulations and Regulatory Guidance

Under Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.90, “Application for amendment of license, construction permit, or early site permit,” whenever a holder of a license wishes to amend the license, including TSs in the license, an application for amendment must be filed, fully describing the changes desired. Under 10 CFR 50.92(a), determinations on whether to grant an applied-for license amendment are to be guided by the considerations that govern the issuance of initial licenses to the extent applicable and appropriate. Both the common standards for licenses in 10 CFR 50.40(a), and those specifically for issuance of operating licenses in 10 CFR 50.57(a)(3), provide that there must be reasonable assurance that the activities at issue will not endanger the health and safety of the public, and that the applicant will comply with the Commission's regulations.

Pursuant to 10 CFR 50.36, “Technical specifications,” TSs for operating reactors are required, in part, to include items in the following five specific categories: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation (LCOs); (3) Surveillance Requirements (SRs); (4) design features; and (5) administrative controls.

Paragraph 50.36(c)(2)(i) of 10 CFR states that LCOs are the lowest functional capability or performance levels of equipment required for safe operation of the facility, and when an LCO of a reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the TSs until the condition can be met.

Paragraph 50.36(c)(3) of 10 CFR requires that TSs include SRs, which are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the LCOs will be met.

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

The TSs for all current PWR licenses require that an SG program be established and implemented to ensure that SG tube integrity is maintained. Programs established by the licensee, including the SG Program, are listed in the administrative controls section of the TSs. All plants have in their TSs a depth-based plugging criterion to remove degraded tubes from service. Some plants, such as Watts Bar, Unit 2, also have alternative repair criteria (ARC) for specific locations in the SGs.

The NRC publishes standard TS (STS) for each operating reactor type as NUREG-series reports. Accordingly, for Westinghouse Electric Company (Westinghouse) plant designs such as Watts Bar, Unit 2, the NRC staff’s review includes consideration of whether the proposed changes are consistent with NUREG-1431, “Standard Technical Specifications, Westinghouse Plants,” Revision 5, Volume 1 and Volume 2 (Reference 12).

In addition, the STS are continuously modified through NRC incorporation of approved generic changes called “travelers,” which are submitted by the industry TSTF. TSTF-510, Revision 2, was approved by the NRC in 2011 as an option for the SG Program TS, and some of the provisions are specific to the SG tube material.

The NRC staff’s guidance for the review of TSs is contained 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” (Reference 13).

2.4 SG Tube Integrity Requirements in the Watts Bar Unit 2 Technical Specifications

For Watts Bar, Unit 2, the requirements for performing SG tube inspections and plugging are in TS 5.7.2.12, “Steam Generator (SG) Program,” while the requirements for reporting the SG tube inspections and plugging are in TS 5.9.9, “Steam Generator Tube Inspection Report.” These TSs currently reflect provisions in TSTF-510, Revision 2, for the original Alloy 600MA tubing.

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. Technical Specification 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. Technical Specification 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 plugging criteria.

The applicable depth-based tube plugging criterion, specified in TS Section 5.7.2.12.c, is 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. The current TSs also include two ARC and one repair method, in TS Sections 5.7.2.12.c.1, 5.7.2.12.c.2, and 5.7.2.12.f.

The TSs related to SG tube and leakage integrity are also contained in TSs 3.4.13 and 3.4.17. Technical Specification 3.4.17, “Steam Generator (SG) Tube Integrity,” requires SG tube integrity to be maintained and tube meeting the tube repair criteria to be plugged in accordance with the SG Program. Technical Specification 3.4.13, “RCS Operational LEAKAGE,” includes a limit on reactor coolant system (RCS) 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 in-service inspection required by the plant TS, the operational leakage limit provides added assurance of timely plant shutdown before tube structural and leakage integrity are impaired, should such a flaw result in primary-to-secondary leakage.

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). 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.

3.0 TECHNICAL EVALUATION

The NRC staff evaluated the licensee's application to determine if the proposed changes are consistent with the guidance, regulations, and plant-specific design and licensing basis information discussed in Sections 2.3 and 2.4 of this safety evaluation.

3.1 Evaluation of Proposed Technical Specification Changes

TVA proposed to delete all tube ARC (i.e., F*, GL 95-05 voltage based, and SG tube sleeve repair) and their bases, as applicable, from the identified places in TSs 3.4.17, 5.7.2.12 and 5.9.9 since these requirements only apply to the original SGs with Alloy 600MA tubes.

TSTF-510, Revision 2, addresses SG inspection and tube sample selection for the three tube materials used in SGs at U.S. plants. The licensee's SG Program in TS 5.7.2.12 is currently based on the design of the original SGs and provisions in TSTF-510, Revision 2, for SGs with Alloy 600MA tubing.

The NRC staff evaluated the proposed TS changes based on the removal of ARC, as well as changes in SG design, and consistency with TSTF-510, Revision 2, for SGs with Alloy 690TT tubing. The staff's evaluation of each proposed change is provided below.

During the review, the NRC staff requested some additional information. Based on the licensee's response in Reference 2, as supplemented by Reference 3, the staff confirmed that the key design parameters for the Watts Bar, Unit 2, RSGs are the same as for the Watts Bar, Unit 1, RSGs. The staff notes the Unit 1 RSGs have operated safely since 2006 with some degradation by support structure wear, but no other degradation such as corrosion or wear from loose parts.

Section 3.4.17, "Steam Generator (SG) Tube Integrity"

The staff finds it acceptable to delete "or repair" and "or repaired" in the LCO, Condition A and associated Required Action A2, and SR 3.4.17.2 because no repairs are approved for the RSGs and the proposed wording is consistent with TSTF-510, Revision 2.

Section 5.7.2.12, "Steam Generator (SG) Program"

- The staff finds it acceptable to delete "or repair" and "or repaired" in TS 5.7.2.12.a because the ARC only applies to the original SGs, and the proposed wording is consistent with TSTF-510, Revision 2.
- The staff finds the proposed wording changes to the accident-induced leakage performance criterion in TS 5.7.2.12.b.2 acceptable because leakage provisions related to the ARC for the original SGs would be deleted and the criterion of 1 gpm in the faulted SG is consistent with TSTF-510, Revision 2.
- The staff finds the proposed change to the TS 5.7.2.12.c acceptable because it deletes the provisions related to repair and sleeves, which no longer apply. The staff notes that the criterion of 40 percent through-wall depth-based plugging is retained.
- The staff finds the changes to TS 5.7.2.12.d acceptable because they would:
    - require inspection of the entire tube length within the tubesheet region, eliminating a region of the tubesheet previously excluded from inspection based on an ARC for the original SGs, and

  - state that the tube-to-tubesheet weld is not part of the tube, using language consistent with TSTF-510, Revision 2
- The staff finds the proposed changes to TS 5.7.2.12.d.2 acceptable because,
  - An inspection period at least every 72 EFPM or every third refueling outage (starting after the first refueling outage) is consistent with TSTF-510, Revision 2, for the tube material (Alloy 690TT) in the Watts Bar Unit 2 RSGs.
  - The proposed definitions of subsequent inspection periods in the new paragraphs (a through d) are consistent with TSTF-510, Revision 2, for the tube material (Alloy 690TT) in the Watts Bar Unit 2 RSGs.
  - The proposed requirements for the minimum number of tubes in all SGs that must be inspected at each scheduled inspection and for potential new degradation mechanisms are consistent with TSTF-510, Revision 2.
- The staff finds it acceptable to delete TS 5.7.2.12.d.4 because it is related exclusively to the voltage-based ARC for the original SGs.
- The staff finds it acceptable to delete TS 5.7.2.12.f because it is related exclusively to the use of sleeves for tube repair in the original SGs.

Section 5.9.9, "Steam Generator Tube Inspection Report"

- The staff finds all proposed changes to TS 5.9.9 acceptable because each one (paragraphs e through h) is related to tube repairs approved in ARC for the original SGs, and because the resulting wording is consistent with TSTF-510, Revision 2.

Conclusion Regarding Proposed Changes to Technical Specifications

Based on the information provided in the LAR, the NRC staff finds the proposed changes to the Watts Bar Unit 2 TS acceptable because they remove TS provisions that apply only to the original SGs, and they are consistent with TSTF-510, Revision 2 for the Alloy 690TT tube material in the RSGs. The staff's basis for concluding TSTF-510, Revision 2, is acceptable is documented in Reference 14. In addition, based on the licensee's response in Reference 2, as supplemented by Reference 3, the staff confirmed that the key design parameters for the Watts Bar, Unit 2, RSGs are the same as for the Watts Bar, Unit 1, RSGs, which have operated safely since 2006.

The NRC staff concludes that the proposed changes to TS 3.4.17, 5.7.2.12, and 5.9.9, are consistent with TSTF-510, Revision 2, for the Watts Bar, Unit 2, RSGs, and meet the requirements of 10 CFR 50.36 as they relate to the Steam Generator Program. Therefore, the NRC staff finds the proposed changes acceptable.

3.2 Evaluation of Proposed Operating License Condition Change

The current Watts Bar, Unit 2, Alloy 600 original SGs are scheduled to be replaced with Alloy 690 RSGs, which are equivalent to those installed in Watts Bar, Unit 1. As a result, the reference to PAD4TCD, which was an interim computer analysis code used to evaluate fuel performance and design, in Watts Bar, Unit 2, Facility Operating License No. NPF-96, Condition 2.C.(4) will no longer apply following installation of the RSGs. Therefore, TVA proposed that the first sentence of License Condition 2.C.(4) be deleted.

In 2018, TVA submitted a LAR (Reference 15) requesting that the NRC allow the use of PAD4TCD to establish core operating limits until the installation of the RSGs. The LAR was approved by the NRC staff (Reference 16). Subsequently, the NRC staff approved a revision to License Condition 2.C.(4) of the Watts Unit 2 (Reference 17) to reflect the implementation of the FULL SPECTRUM™ Loss-of-Coolant Accident (FSLOCA™) Evaluation Methodology (EM), which will be implemented when the Watts Bar, Unit 2, SGs are replaced. This portion of the license condition will remain after the SGs are replaced.

In Section 3.3.6 of Reference 17, the NRC staff noted that PAD5 fuel performance data was utilized in the Watts Bar, Units 1 and 2, analysis with the FSLOCA EM, and the staff concluded in its safety evaluation that given that the licensee used the latest NRC-approved fuel performance code (i.e., PAD5) and used appropriate conservative inputs, the licensee met the requirements of the limitation and condition of the FSLOCA EM. Therefore, the proposed change to License Condition 2.C.(4) supports the TVA plan for transitioning to PAD5, the latest fuel performance model approved for use by the NRC (Reference 18), after installation of the replacement SGs. The NRC staff, therefore, finds the proposed change to the Facility Operating License No. NPF-96, License Condition 2.C.(4) 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 August 2, 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 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 July 13, 2021 (86 FR 36781). 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. Polickoski, J. T., TVA, letter to U.S. NRC, "Application to Modify Watts Bar Nuclear Plant, Unit 2 Technical Specifications Steam Generator Inspection/Repair Program Provisions and Unit 2 Facility Operating License Condition 2.C.(4) (WBN-TS-20-06)," March 11, 2021 (ADAMS Accession No. ML21070A432).

2. Polickoski, J. T., TVA, letter to U.S. NRC, "Response to Request for Additional Information Regarding Application to Modify Watts Bar Nuclear Plant, Unit 2 Technical Specifications Steam Generator Inspection/Repair Program Provisions and Unit 2 Facility Operating License Condition 2.C(4) (WBN-TS-20-06)," August 11, 2021 (ADAMS Accession No. ML21223A319).

3. Polickoski, J. T., TVA, letter to U.S. NRC, "Supplement to Response to Request for Additional Information Regarding Application to Modify Watts Bar Nuclear Plant, Unit 2 Technical Specifications Steam Generator Inspection/Repair Program Provisions and Unit 2 Facility Operating License Condition 2.C(4) (WBN-TS-20-06)," January 5, 2022 (ADAMS Accession No. ML22005A344).

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

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).

6. Watts Bar Nuclear Plant, Dual-Unit Updated Final Safety Analysis Report, Amendment 3, Chapter 3, "Design of Structures, Components, Equipment, and Systems," October 29, 2020 (ADAMS Accession No ML20323A314).

7. Inspection Procedure 50001, "Steam Generator Replacement Inspection," September 12, 2017 (ADAMS Accession No. ML17083B311).

8. Inspection Procedure 71111.17T, "Evaluations of Changes, Tests, and Experiments," December 8, 2016 (ADAMS Accession No. ML16340A998).

9. Inspection Procedure 71111.18, "Plant Modifications," March 31, 2021 (ADAMS Accession No. ML21040A185).

10. Inspection Procedure 71111.21M, "Design Bases Assurance Inspection (Team)," December 8, 2016 (ADAMS Accession No. ML16340B000).

11. NRC Generic Letter 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking," August 3, 1995 (ADAMS Accession No. ML19355B268).

12. NUREG-1431, "Standard Technical Specifications, Westinghouse Plants," Revision 5, Volumes 1 and 2, September 2021 (ADAMS Accession Nos. ML21259A155 and ML21259A159).

13. 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, March 2010 (ADAMS Accession No. ML100351425).

14. Model Safety Evaluation for Plant-Specific Adoption of Technical Specifications Task Force Traveler TSTF-510, Revision 2, "Revision to Steam Generator Inspection Program Inspection Frequencies and Tube Sample Section," using the Consolidated Line Item Improvement Process, October 19, 2011 (ADAMS Accession No. ML112101513).

15. Shea, J. W., TVA, letter to U.S. NRC, CNL-18-016, "Watts Bar Nuclear Plant Unit 2 - Application to Revise License Condition 2.C(4) PAD4TCD (391-WBN-TS-18-03)," March 5, 2018 (ADAMS Accession No. ML18064A192).

16. Schaaf, R. G., U.S. NRC, letter to Shea, J. W., TVA, "Watts Bar, Unit 2 – Issuance of Amendment Regarding Application to Revise License Condition 2.C.(4) PAD4TCD (EPID L-2018-LLA-0051)," March 20, 2019 (ADAMS Accession No. ML19046A286).

17. Green, K. J., NRC letter to Barstow, J., TVA, "Watts Bar Nuclear Plant, Units 1 and 2 - Issuance of Amendment Nos. 143 and 50 Regarding Implementation of FULL SPECTRUM™ Loss-of-Coolant Accident Analysis (LOCA) and New LOCA-Specific Tritium Producing Burnable Absorber Rod Stress Analysis Methodology (EPID L-2020-LLA-0005)," February 26, 2021 (ADAMS Accession No. ML21034A169).

18. Westinghouse Electric Company report WCAP-17642-A, "Westinghouse Performance Analysis and Design Model (PAD5), Revision 1," November 2017 (ADAMS Accession No. ML17338A396).

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Date: January 25, 2022

SUBJECT: WATTS BAR NUCLEAR PLANT, UNIT 2 - ISSUANCE OF AMENDMENT NO. 60 REGARDING REVISION OF TECHNICAL SPECIFICATION REQUIREMENTS SPECIFIC TO THE MODEL D3 STEAM GENERATORS THAT WILL NO LONGER APPLY FOLLOWING STEAM GENERATOR REPLACEMENT (EPID L-2021-LLA-0043) DATED JANUARY 25, 2022

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