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

June 24, 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 2 - ISSUANCE OF AMENDMENT NO. 54 REGARDING USE OF TEMPERATURE ADJUSTMENT TO VOLTAGE GROWTH RATE FOR THE GENERIC LETTER 95-05 STEAM GENERATOR TUBE REPAIR CRITERIA (EPID L-2021-LLA-0026)

Dear Mr. Barstow:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 54 to Facility Operating License No. NPF-96 for the Watts Bar Nuclear Plant (Watts Bar), Unit 2. This amendment is in response to your application dated February 25, 2021, as supplemented by letters dated March 23 and May 14, 2021.

This amendment revises the Watts Bar Updated Final Safety Analysis Report to apply a temperature adjustment to the voltage growth rate calculation used to determine the end-of-cycle distribution of indications of axial outer diameter stress corrosion cracking at tube support plates in support of the Watts Bar, Unit 2, operational assessment.

A copy of our 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. 54 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. 54
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 February 25, 2021, as supplemented by letters dated March 23, and May 14, 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, by Amendment No. 54, the license is amended to authorize revision to the Updated Final Safety Analysis Report (UFSAR), as set forth in the application dated February 25, 2021, as supplemented by letters dated March 23 and May 14, 2021. The licensee shall update the UFSAR to incorporate the temperature adjusted growth rate calculation as described in the licensee's application, as supplemented, and the NRC staff's safety evaluation attached to this amendment.

3. This license amendment is effective as of the date of its issuance, and shall be implemented by August 1, 2021. The UFSAR changes shall be implemented in the next periodic update to the UFSAR in accordance with 10 CFR 50.71(e).

FOR THE NUCLEAR REGULATORY COMMISSION

David J. Wrona

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Date: 2021.06.24 13:08:56 -04'00'

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

Date of Issuance: June 24, 2021

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 54 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 February 25, 2021 (Reference 1), as supplemented by letters dated March 23, 2021 (Reference 2), and May 14, 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), dual-unit Updated Final Safety Analysis Report (UFSAR). The requested changes would revise UFSAR Section 5.5.2.4, "Test and Inspections," for Watts Bar, Unit 2 only, to apply a temperature adjustment to the voltage growth rate calculation used to determine the end-of-cycle distribution of indications of axial outer diameter stress corrosion cracking (ODSCC) at tube support plates (TSPs) in support of the Watts Bar, Unit 2, operational assessment (OA) for its steam generators (SGs), and a conforming change to add two new references to the reference section in UFSAR Section 5.5.

The supplements dated March 23 and May 3, 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, Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on March 24, 2021 (86 FR 15727).

2.0 REGULATORY EVALUATION

2.1 System Description

Watts Bar, Unit 2 has four Westinghouse Model D3 SGs, each of which contains 4,674 mill-annealed Alloy 600 tubes. Each tube has a nominal outside diameter of 0.75 inches and a nominal wall thickness of 0.043 inches. The tubes are supported by anti-vibration bars (AVBs) and carbon steel drilled tube support plates (TSPs). Each SG has an integral preheater section with flow distribution baffle plates. The SGs are further described in Watts Bar UFSAR Section 5.5.2.2 (Reference 4).

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.

2.2 Licensee's Requested Changes

For Cycle 4a[1], the licensee reduced the Watts Bar, Unit 2, operating temperature (as expressed by the hot-leg temperature, T_{hot}) in order to reduce the growth rate of SCC in the SG tubes. This was based on the understanding that the rates of many physical and chemical processes increase with increasing temperature according to a relationship described mathematically by the Arrhenius equation. The equation is described in more detail Section 3.2.1 of this safety evaluation. The licensee would apply the temperature reduction to the GL 95-05 voltage growth rate calculations and corresponding EOC 4a voltage distributions for axial ODSCC at TSP intersections.

The licensee requested approval of the temperature adjustment as a revision to Section 5.5.2.4 of the dual-unit UFSAR for Unit 2 only, and proposed conforming changes to the reference section of Section 5.5 of the UFSAR, as described below.

Proposed Addition to Section 5.5.2.4, Tests and Inspections

> Also, when normal operating temperature differences exist from either cycle-to-cycle, or within a cycle, an exception to the GL 95-05 analysis in the form of a temperature adjustment to the growth rate calculation in accordance with Section 10.5.6.1.6 of Reference 27 will be applied. The temperature adjustment methodology will be used to determine the End of Cycle voltage distribution of axial indications for comparison to the conditional probability of tube burst of less than or equal to 1×10^{-2} and to determine the total primary-to-secondary leak rate from an affected SG during a postulated main steam line break event. The upper voltage repair limit will be determined using the guidance of GL 95-05 and the plant-specific average growth rate will correspond to the temperature applicable to 100% reactor power operation. This exception applies until the Unit 2 Steam Generators are replaced(28).

Proposed Addition to Section 5.5, References

> 27. EPRI Report 1018047, "Steam Generator Tubing Outside Diameter Stress Corrosion Cracking at Tube Support Plates Database for Alternate Repair Limits: Addendum 7," EPRI, dated September 2008.
>
> 28. NRC letter to TVA, "XXXXX (TAC No. XXXX)," dated MM/DD/YY (MLXXXX) (Thot LAR)

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[1] In Reference 3, the licensee defined Cycle 4a for the GL 95-05 OA as the period of operation from entering Mode 4 on November 16, 2020, following the U2R3 refueling outage, until the start of the mid-cycle SG inspection outage (no later than September 11, 2021). Cycle 4b is defined as the period of operation following the mid-cycle inspection outage until the start of the U2R4 refueling outage scheduled for March 2022.

2.3 Regulatory Requirements and Guidance

The NRC staff considered the following regulations and guidance during its review of this amendment.

Section 50.34, "Contents of applications; technical information," of Title 10 of the *Code of Federal Regulations* (10 CFR) contains the requirements for the final safety analysis report (FSAR). In part, 10 CFR 50.34(b) requires each application for an operating license to include an FSAR that describes the facility, presents the design bases and limits on its operation, and presents a safety analysis of the structures, systems, and components and of the facility as a whole. Section 50.34(b)(6)(iv) requires plans for conduct of normal operations, including maintenance, surveillance, and periodic testing of structures, systems, and components.

Fundamental regulatory requirements with respect to the integrity of the SG tubing are established in 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 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).

The 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. The Watts Bar construction permit was issued in January, 1973. The Watts Bar UFSAR, however, 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, as documented in UFSAR Section 3.1, "Conformance with the NRC General Design Criteria" (Reference 5).

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 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 technical specifications (TSs).

In 10 CFR 50.36, "Technical specifications," the regulatory requirements related to the content of the TSs are established. Per 10 CFR 50.36(b), each license authorizing operation of a utilization facility will include technical specifications. 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 to operate the facility in a safe manner. All plants have in their TSs a depth-based plugging criterion to remove degraded tubes from service. Some plants also have alternative repair criteria (ARC) for specific locations in the SGs.

Generic Letter (GL) 95-05 provides guidance to licensees who may wish to request a license amendment to the plant technical specifications to implement alternate steam generator tube repair criteria applicable specifically to ODSCC at the tube-to-tube support plate intersections in Westinghouse-designed steam generators having drilled-hole TSPs and alloy 600 tubing (Reference 6). These criteria do not set limits on the depth of ODSCC indications to ensure tube integrity margins; instead, it relies on correlating the eddy current voltage amplitude from a bobbin coil probe with the more specific measurement of burst pressure and leak rate. The NRC staff recognizes that although total margin may be reduced following application of the voltage-based repair guidance of this GL, this guidance does ensure that structural and leakage integrity continues to be maintained with an acceptable level of margin consistent with GDCs 14, 15, 30, 31, and 32 of 10 CFR Part 50, Appendix A and the limits of 10 CFR Part 100. Since the voltage-based repair criteria do not incorporate minimum wall thickness requirements, there is a possibility that tubes with up to 100 percent through-wall cracks can remain in service. The staff included provisions for augmented SG tube inspections and restrictive operational leakage limits because of the increased likelihood of such flaws.

2.4 Steam Generator 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.” 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 5.7.2.12.c.1, 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. Watts Bar, Unit 2, also has an ARC specified in TS 5.7.2.12.c.2 to apply the criteria for voltage-based plugging limits for axial ODSCC in accordance with the guidance in GL 95-05. The NRC reviewed and approved the use of this ARC at Watts Bar, Unit 2, on June 3, 2019 (Reference 7). The licensee implemented the GL 95-05 ARC at Watts Bar, Unit 2, for the first time during the fall 2020 refueling outage (U2R3). The NRC subsequently reviewed and approved the use of alternate probability of detection (POD) values for the GL 95-05 ARC at Watts Bar, Unit 2, on February 9, 2021 (Reference 8). Technical Specification 5.9.9 includes reporting criteria specific to the GL 95-05 ARC.

Technical specifications 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.

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 accident analyses are described in Chapter 15 of the Watts Bar UFSAR (Reference 9). No accident analyses for Watts Bar, Unit 2, are being changed because of the proposed amendment and, thus, no radiological consequences of any accident analysis are being changed. The proposed changes maintain the accident analyses and consequences that the NRC has reviewed and approved for the postulated design-basis accidents for SG tubes.

3.0 TECHNICAL EVALUATION

3.1 Background

3.1.1 Inspection and Degradation Summary

Watts Bar, Unit 2, began commercial operation in October 2016. Inservice inspections of the SGs were performed in fall 2017 (U2R1), spring 2019 (U2R2), and fall 2020 (U2R3). The inspection scope and results for each inspection is summarized in more detail in Reference 8. In the U2R1 inspection in fall 2017 (Reference 10), primary-side eddy current inspections were performed over the full length of the in-service tubes in each SG using a combination of bobbin probe, array probe, and +Point™ rotating probe. The only degradation mechanisms detected in U2R1 were wear at TSPs and AVBs, but no tubes were plugged because of wear.

In the U2R2 inspection in spring 2019 (Reference 11), the primary-side eddy current inspections were the same as in the U2R1 outage. The following degradation mechanisms were detected: wear at tube support structures (both TSPs and AVBs), circumferential ODSCC at the hot-leg top-of-tubesheet (TTS), and axial ODSCC at TSPs. With respect to axial ODSCC at TSPs, which is the degradation mechanism addressed by GL 95-05, eight indications were detected in eight different tubes. Because the GL 95-05 ARC was not yet implemented during the U2R2 outage, all eight tubes were plugged. In addition, five tubes were plugged due to circumferential ODSCC at the hot-leg TTS.

In the U2R3 inspection in fall 2020 (Reference 12), the scope of the inspections was similar to the first two inspections. The primary-side eddy current inspections were conducted with a combination of bobbin probes, array probes, and +Point™ rotating probes. The amount of degradation in the SG tubes increased during the third operating cycle. The degradation mechanisms detected in the U2R3 inspections included the previously detected mechanisms – wear from tube support structures, circumferential ODSCC at the hot-leg TTS, and axial ODSCC at TSPs – as well as stress corrosion cracking (SCC) mechanisms detected for the first time. The newly detected mechanisms were axial ODSCC at the hot-leg TTS, axial primary water stress corrosion cracking at the hot-leg TTS, circumferential ODSCC at freespan dings, and axial ODSCC at freespan dings. The total number of tubes plugged due to SCC mechanisms was 188.

The GL 95-05 ARC is applicable to most of the ODSCC indications detected in U2R3 at the TSPs and FDB, and the ARC was implemented to allow the tubes with those indications to remain in service if they meet the GL 95-05 acceptance criteria.

3.1.2 Generic Letter 95-05 Application to the Fall 2020 Inspection

The NRC approved the use of the GL 95-05 methodology for Watts Bar, Unit 2, during Cycle 3 (Reference 7). Based on the results of each SG inspection, the licensee calculates conditional probability of burst (POB) and accident-induced leakage (AIL) to determine if the acceptance limits were met during the just-completed operating cycle (condition monitoring, or CM), and to ensure they will be met during the next operating cycle (operational assessment, or OA). To relate voltage to burst and leakage, the licensee uses the empirically derived correlation in WCAP-14277, Revision 1 (Reference 13).

When the licensee first applied the GL 95-05 criteria following the U2R3 inspection, including the default POD of 0.6 for all indications, the calculated conditional burst probability exceeded the $1x10^{-2}$ performance criterion for one of the SGs (SG 3). The conditional burst probability is based on the bobbin probe voltage for the axial ODSCC indications at TSP intersections. Indications with voltages higher than the upper repair limit caused the SG 3 conditional burst probability to exceed $1x10^{-2}$ at EOC 3. The GL 95-05 criteria include for each inspection a determination of the upper voltage repair limit, which allows for measurement uncertainty and flaw growth during the subsequent operating cycle. The licensee reported there were 12 indications with voltage greater than or equal to the U2R3 2.85-volt upper repair limit (Reference 14). All the tubes with indications exceeding the upper voltage repair limit were plugged during the outage.

Although the tubes with voltage indications greater than the upper voltage repair limit were plugged in accordance with GL 95-05, the indications were included in calculating the distribution of indications at the beginning of Cycle 4. This distribution is used for projecting the EOC 4 voltage distribution, conditional burst probability, and leakage. Using a default POD of 0.6 in the projections, the conditional burst probability criterion was met for only a short period of operation. In Reference 8, the NRC-approved voltage-dependent alternate POD values for Watts Bar, Unit 2, beginning with the U2R3 inspection, that recognized large voltage indications have much higher detection probabilities. The alternate POD values are 0.6 for indications with bobbin voltage less than 3.2 volts, 0.90 for indications greater than or equal to 3.2 volts and less than 6.0 volts, and 0.95 for indications 6.0 volts and greater. Using these alternative voltage-dependent POD values, the licensee concluded that both the conditional burst probability and leakage performance criteria would be met for an operating interval of up to 285 calendar days at full power, at which time a mid-cycle inspection would be required (Reference 14). To extend the operating time before the mid-cycle outage, the licensee is operating at a reduced temperature to reduce the rate of stress corrosion cracking. Although the benefit of reducing temperature on stress corrosion crack growth has been demonstrated in testing and operating experience, the GL 95-05 generic methodology does not include provisions for a temperature adjustment. Therefore, the licensee requested a license amendment. The proposed changes in the amendment and staff review are discussed below.

3.2 Staff Evaluation of Proposed UFSAR Changes

3.2.1 Basis for a Temperature Adjustment

In response to tube cracking at support plate intersections, the licensee is operating at reduced power, with a corresponding lower T_{hot}. The lower temperature is intended to reduce the rate of degradation and extend the operating time, relative to full-power operation, until the next required SG tube inspection. Tube integrity is defined in the TS in terms of structural integrity and leakage integrity applicable to any form of tube degradation. The licensee's proposal is specific to degradation by ODSCC of the tubes at intersections with the TSPs and FDB, which is managed according to the alternate repair criteria in GL 95-05. Continuing to meet the GL 95-05 burst and leakage criteria will assure that tube integrity is maintained. Because operating temperature is not addressed in the GL 95-05 ARC, as implemented by Watts Bar, Unit 2, the licensee identified this as an exception to GL 95-05 and proposed a change to the UFSAR to document how operation of the plant at a reduced temperature is used to maintain tube integrity until the next inspection.

The Arrhenius equation describes the temperature dependence of the rate of many physical and chemical processes, including many corrosion processes. References 15 and 16 are examples of this behavior for SCC of Alloy 600MA in laboratory tests in simulated reactor coolant environments from about 290 to 360 degrees Celsius (C) (554 to 680 degrees Fahrenheit (F)), which bounds the reactor coolant temperature for operating commercial PWRs. The general form of the Arrhenius equation is shown below, where A is a constant, Q is the activation energy, R is the ideal gas constant, and T is the absolute temperature (Reference 17). Since an activation energy represents a barrier that must be overcome for the process to occur, activation energy is always positive. Therefore, the rate will increase with increasing temperature for processes that follow the Arrhenius equation.

$$Rate = Ae^{\frac{-Q}{RT}}$$

Reference 18 provides the EPRI database for industry experience with ODSCC at TSPs, including eddy current inspection results, destructive evaluation results for tubes pulled from SGs, and recommended analysis methods for applying the GL 95-05 ARC. This database, which is referenced by the licensee in the proposed UFSAR revision, presents the Arrhenius equation as a way to adjust voltage growth for temperature and a way to normalize data to compare plants operating at different temperatures. This method of normalizing degradation growth rates with the Arrhenius equation is also included in the industry guidelines for assessing SG tube integrity (Reference 19). Voltage growth is the only GL 95-05 parameter for which temperature adjustment is proposed.

The temperature adjustments in the EPRI database (Reference 18) are contained in Sections 10.5.6.1.4 and 10.5.6.1.6 of that document. Section 10.5.6.1.4, "Temperature of Operation," provides an expression for converting the time at one temperature to the time for an equivalent amount of crack growth at another temperature. Section 10.5.6.1.6, "Adjustment of Voltage Growth," provides a form of the same expression in terms of voltage change as a function of temperature and operating time. The licensee is proposing to use the latter for Watts Bar, Unit 2, to calculate the amount of operating time until crack growth causes the POB or AIL acceptance criteria to be exceeded. Although there is a technical basis for adjusting the voltage growth rate for temperature in this way, the NRC staff does not review or approve the EPRI database or industry guidelines, and there is no precedent for a licensee adjusting its GL 95-05 analysis results based on changes in the plant operating temperature. Therefore, using the

proposed temperature adjustment is an exception in the application of the GL 95-05 ARC approved for Watts Bar, Unit 2.

3.2.2 Staff Evaluation of Proposed Temperature Adjustment Methodology

GL 95-05 Voltage Growth Description

Generic Letter 95-05 contains criteria for determining flaw voltage distributions and voltage growth rates from crack growth, and for using these growth rates to project the EOC flaw voltage distributions. For the first application of the GL 95-05 ARC at Watts Bar, Unit 2, in fall 2020, the licensee provided the voltage growth-rate distributions and projected EOC 4a voltage distributions for each SG in Reference 12. Based on these projections, the licensee determined the conditional POB and accident leak rate criteria could be met for a Cycle 4a period of up to 285 days.

As stated in Reference 1, the licensee is currently operating Watts Bar, Unit 2, at a reduced temperature relative to Cycle 3 to slow the progression of axial ODSCC at TSP intersections. The temperature reduction is approximately 4 degrees Fahrenheit in the primary system reactor vessel outlet temperature (T_{hot}). To justify an increase in the Cycle 4a period beyond 285 days, the licensee proposed using the Arrhenius relationship in Section 10.5.6.1.6 in Reference 18 to adjust the flaw growth. This form of the Arrhenius equation relates the voltage growth (ΔV) for one temperature (T) and period (t) to the voltage growth for a different temperature and period. This adjustment would then be applied to the growth distribution to determine the reduced growth rate and corresponding increased operating period. This approach assumes growth is a linear function of the length of the operating cycle, which is consistent with GL 95-05.

$$\Delta V_2 = \Delta V_1 \frac{\Delta t_2}{\Delta t_1} e^{\frac{-Q}{R}\left(\frac{1}{T_2} - \frac{1}{T_1}\right)}$$

The licensee stated in Reference 3 that only one temperature difference will be applied during Cycle 4a, and possibly one temperature difference during Cycle 4b. The temperature difference, if any, for Cycle 4b will depend on the mid-cycle inspection results. The NRC staff evaluated the licensee's request in that context, considering operation at one reduced temperature compared to operating at the Cycle 3 temperature. The staff did not evaluate temperature adjustments to any other parameters. In response to an NRC staff request for additional information (Reference 3), the licensee confirmed they were no longer pursuing the original proposal to apply a temperature adjustment to the GL 95-05 upper voltage repair limit.

Applicability of the Arrhenius Equation

In addition to the references provided in the LAR, the NRC staff reviewed literature for confirmation that the rate of the axial ODSCC at TSPs in Alloy 600MA SG tubing follows an Arrhenius temperature dependence. Reference 15 presents plots of log crack growth rate (CGR) versus temperature for Alloy 600MA tubing in simulated reactor coolant from 290 to 365 degrees C (554 to 689 degrees F). The plots show linear behavior for all stress conditions and water chemistry variations studied, which is characteristic of Arrhenius behavior, with the slope representing the activation energy. Reference 16 includes plots from multiple sources showing the same type of CGR versus temperature relationship for Alloy 600 weld alloys (as-welded Alloy 82 and Alloy 182) in simulated reactor coolant from 290 to 360 degrees C (554 to 680 degrees F). Based on References 15, 16, and 18, the temperature response

observed in Alloy 600 materials, including mill-annealed SG tubing, over a temperature range that bounds the Watts Bar, Unit 2, operating temperature range, the NRC staff finds that applying the Arrhenius temperature adjustment is technically justified and acceptable for the Watts Bar, Unit 2, GL 95-05 voltage growth rates.

Activation Energy

The activation energy value for crack growth has a significant effect on the licensee's proposed temperature adjustment. The Arrhenius relationship results in a linear plot of log CGR versus temperature, where the slope of the line is the activation energy. The type of CGR tests described in References 15 and 16 have been used to derive activation energies as a function of variables such as material composition, heat treatment, and test conditions. Activation energy values determined this way have an uncertainty according to the linear fit to the test data. In Enclosure 3 of Reference 1, the licensee proposed a value of 30 kcal/mol for Watts Bar, Unit 2, based, in part, on the recommendations in the EPRI database (Reference 18) and EPRI Integrity Assessment Guidelines (Reference 19). Similar values have been reported in other sources. For example, Reference 15 reported a value of 33 kcal/mol for samples fabricated from Alloy 600MA tubing in both the as-received and cold-worked conditions. A report from NRC-sponsored research refers to 31 kcal/mol as the a commonly used value for crack growth in thick sections of Alloy 600 (Reference 16).

According to the Arrhenius equation, higher activation energy results in a greater effect of temperature on the CGR and adjusted operating period. For example, for a temperature decrease from 617 to 613 degrees F applied to a baseline operating period of 285 days, the adjusted operating periods are 313, 314, and 316 days, respectively, for activation energy values of 30, 31, and 33 kcal/mol. The baseline operating period and temperature values are from References 1 and 2. The NRC staff performed this example calculation based on the following expression from Section 10.5.6.1.4 in Reference 18, derived from the Arrhenius equation, where t_1 and t_2 are the times producing the same amount of flaw growth at two different temperatures T_1 and T_2:

$$t_2 = t_1 \, e^{\frac{Q}{R}\left(\frac{1}{T_2} - \frac{1}{T_1}\right)}$$

This equation and examples above show that a lower activation energy (Q) results in a smaller increase in the adjusted operating period, which is conservative in terms of crediting the lower operating temperature with reducing flaw growth rate. The licensee used an activation energy of 30 kcal/mol for the proposed temperature adjustment, which is the lowest of the measured and recommended values described above. The NRC staff finds the licensee's use of 30 kcal/mol acceptable because it is consistent with the recommendations in the large industry database and guidelines for ODSCC in Alloy 600MA SG tubing and is appropriate considering other values found in the literature.

Application to Cycle 4a

The licensee's LAR submittal (Reference 1) includes a graph illustrating the effect of the proposed Cycle 4a temperature adjustment (4 degrees F) on the limiting voltage growth distribution from Cycle 3 (SG 3). The adjusted growth distribution reflects a reduction factor of 0.91 calculated from the Arrhenius relationship, with a corresponding increase in the operating period of 27 calendar days at current operating conditions (312 calendar days total). Reference 1 did not identify the temperatures used in the adjustment; however, the NRC staff

calculated the same reduction factor of 0.91 using temperature values of 617 degrees F (T_1) and 613 degrees F (T_2) based on the highest loop average T_{hot} during Cycle 3 identified in the U2R3 OA (Reference 2).

In addition to the temperature adjustment calculation, the NRC staff considered the margin between the planned operating period for Cycle 4a and the operating period supported by the OA. The licensee stated that the proposed temperature adjustment for the GL 95-05 ARC would enable an operating period of 312 days before exceeding the tube integrity performance criteria. The licensee also stated that the mid-cycle inspection outage is scheduled to begin on September 11, 2021, for an operating period of 299 days, leaving a margin of 13 days (Reference 3). For the other SCC and tube wear mechanisms, the licensee's OA concluded that the performance criteria would be met for operation through Cycles 4a and 4b (1.38 EFPY total), including an assumption of a return to the Cycle 3 T_{hot} of 617 degrees F during Cycle 4b (Reference 2). The licensee also stated that the use of 4 degrees F in the temperature adjustment calculation is conservative because the actual temperature reduction is greater than 5 degrees F in the hottest reactor coolant loop.

The NRC staff finds the licensee's GL 95-05 temperature adjustment for Cycle 4a acceptable because it provides reasonable assurance that the tube burst and leakage criteria will be met with margin until the mid-cycle tube inspections. This includes the staff's evaluation of the applicability of the Arrhenius equation and activation energy as discussed in the previous sections, as well as the licensee's OA showing that the burst and leakage criteria will be met for a longer operating period for the other degradation mechanisms. The NRC staff's calculations confirmed the licensee's Cycle 4a calculated adjusted operating period of an additional 27 days. The licensee's planned start date for the mid-cycle outage on September 11, 2021, represents a 14-day adjusted operating period, leaving 13 days of margin. If the licensee changes the start date of the mid-cycle outage, it should ensure that sufficient margin remains in the conditional burst probability and leakage criteria.

Application to Cycle 4b

The proposed UFSAR changes would allow the licensee to apply a single temperature adjustment to Cycle 4b following the mid-cycle inspection, potentially for a different temperature than for Cycle 4a. The NRC staff finds this acceptable and notes that the mid-cycle inspection results for all degradation mechanisms, condition monitoring, and operational assessment will determine the appropriate temperature adjustment, if any, to Cycle 4b. In addition, should a greater temperature reduction be warranted, the licensee will need to confirm other analyses affected by RCS temperature, such as UFSAR Chapter 15 accident analyses, remain valid.

NRC Evaluation Summary of the Proposed FSAR Changes

Based on the information provided in the LAR, the NRC staff finds the proposed temperature adjustment to the ODSCC voltage growth rate (GL 95-05 ARC) and corresponding UFSAR changes acceptable because: (a) the Arrhenius temperature relationship is applicable to ODSCC voltage growth in the Watts Bar, Unit 2, SG tubes evaluated using GL 95-05; (b) the activation energy assumed in the calculations is consistent with industry guidelines and supported by testing; (c) the licensee included margin in the applied temperature reduction and in the timing of the planned mid-cycle outage; and (d) the proposed changes do not affect the other provisions in the GL 95-05 criteria for conservatively addressing uncertainties. This temperature adjustment is only applicable until the SGs are replaced. Therefore, the staff finds that the licensee has demonstrated that the structural and leakage integrity of the Watts Bar,

Unit 2, SG tubes will continue to be met. Additionally, the staff finds that the licensee has provided an acceptable description of its operating and surveillance plans, as they relate to the Watts Bar Unit 2 SG tubes, in accordance with 10 CFR 50.34(b)(6)(iv).

3.3 Technical Evaluation Conclusion

The NRC staff concludes that the information submitted by the licensee provides reasonable assurance that the Watts Bar, Unit 2, SG tubes will continue to maintain structural and leakage integrity if the licensee's proposed temperature adjustment is applied to voltage growth in tubes evaluated using the GL 95-05 alternate repair criteria. The staff finds that the licensee's inclusion of the proposed temperature adjustment to the voltage growth rate calculation in the UFSAR meets the requirements of 10 CFR 50.34(b)(6)(iv). Therefore, the NRC staff finds it acceptable for the licensee to apply the proposed temperature adjustment to the GL 95-05 ARC and to make the proposed UFSAR revisions.

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 previously issued a proposed finding that the amendment involves no significant hazards consideration published in the *Federal Register* on March 24, 2021 (86 FR 15727), and there has been no public comment on such finding. 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

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Date: June 24, 2021

SUBJECT: WATTS BAR NUCLEAR PLANT, UNIT 2 - ISSUANCE OF AMENDMENT NO. 54 REGARDING USE OF TEMPERATURE ADJUSTMENT TO VOLTAGE GROWTH RATE FOR THE GENERIC LETTER 95-05 STEAM GENERATOR TUBE REPAIR CRITERIA (EPID L-2021-LLA-0026) DATED JUNE 24, 2021

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