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

September 6, 2016

Mr. Joseph W. Shea
Vice President, Nuclear Licensing
Tennessee Valley Authority
1101 Market Street, LP 3R-C
Chattanooga, TN 37402-2801

SUBJECT: WATTS BAR NUCLEAR PLANT, UNIT 2 – ISSUANCE OF AMENDMENT REGARDING REVISED STEAM GENERATOR INSPECTION SCOPE USING THE F* METHODOLOGY (CAC NO. MF7218)

Dear Mr. Shea:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 2 to Facility Operating License No. NPF-96 for the Watts Bar Nuclear Plant, Unit 2. This amendment consists of changes to the license in response to your application dated December 15, 2015, as supplemented by letters dated May 4, 2016, and June 1, 2016.

The amendment revises the Technical Specifications to allow implementation of the F* (F-star) alternate repair criterion for steam generator tubes.

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

If you have any questions regarding this letter, please contact me at (301) 415-6020 or Robert.Schaaf@nrc.gov.

Sincerely,

Robert G. Schaaf, 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. 2 to NPF-96
2. Safety Evaluation

cc w/enclosures: Distribution via 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. 2
License No. NPF-96

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

   A. The application for amendment by the Tennessee Valley Authority (TVA or the licensee) dated December 15, 2015, as supplemented by letters dated May 4, 2016, and June 1, 2016, 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.

Enclosure 1

2. Accordingly, the Facility Operating License No. NPF-96 is amended as indicated in the attachment to this license amendment.

   Paragraph 2.C.(2) 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. 2 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated Into this license. TVA shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance, and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Tracy J. Orf

Tracy J. Orf, Acting Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Facility Operating
License and Technical Specifications

Date of Issuance: September 6, 2016

ATTACHMENT TO LICENSE AMENDMENT NO. 2

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 page 3.

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	INSERT
5.0-16	5.0-16
5.0-17	5.0-17
--	5.0-17a

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

(1) Maximum Power Level

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

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A as revised through Amendment No. 2 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 February 1, 2017.

(4) PAD4TCD may be used to establish core operating limits for Cycles 1 and 2 only. PAD4TCD may not be used to establish core operating limits for subsequent reload cycles.

(5) By December 31, 2017, 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.

(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:

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 is not to exceed 1 gpm per 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.

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

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

(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 1.64 inches below the bottom of the roll transition or 1.64 inches below the top of the tubesheet, whichever is lower at the tube inlet, to 1.64 inches below the bottom of the roll transition or 1.64 inches below the top of the tubesheet, whichever is lower at the tube outlet, and that may satisfy the applicable tube plugging criteria. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. A degradation assessment shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

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

2. After the first refueling outage following SG installation, inspect each SG at least every 24 effective full power months or at least every refueling outage (whichever results in more frequent inspections). In addition, inspect 100% of the tubes at sequential periods of 60 effective full power months beginning after the first refueling outage inspection following SG installation. Each 60 effective full power month inspection period may be extended up to 3 effective full power months to include a SG inspection outage in an inspection period and the subsequent inspection period begins at the conclusion of the included SG inspection outage. If a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at this location and that may satisfy the applicable tube plugging criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated.

(continued)

5.7.2.12 Steam Generator (SG) Program (continued)

The fraction of locations to be inspected for this potential type of degradation at this location at the end of the inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the inspection period after the determination that a new form of degradation could potentially be occurring at this location divided by the total number of times the SG is scheduled to be inspected in the inspection period.

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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 2 TO FACILITY OPERATING LICENSE NO. NPF-96

TENNESSEE VALLEY AUTHORITY

WATTS BAR NUCLEAR PLANT, UNIT 2

DOCKET NO. 50-391

1.0 INTRODUCTION

By application dated December 15, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15362A023), as supplemented by letters dated May 4, 2016, and June 1, 2016 (ADAMS Accession Nos. ML16127A232 and ML16158A385, respectively), Tennessee Valley Authority (the licensee) requested changes to the Technical Specifications (TSs) for the Watts Bar Nuclear Plant (WBN), Unit 2. The supplements dated May 4,2016, and June 1, 2016, 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 February 16, 2016 (81 FR 7844).

The proposed changes would allow implementation of the F* (pronounced F-star) alternate repair criterion for steam generator (SG) tubes. The F* criterion no longer requires inspection of the lower portion of the tube within the tubesheet, thereby allowing any flaws to remain in service, provided the upper portion of the tube within the tubesheet is free of service-induced flaws.

2.0 REGULATORY EVALUATION

The SG tubes function as an integral part of the reactor coolant pressure boundary and isolate radiological fission products in the primary coolant from the secondary coolant and the environment. Because of the importance of SG tube integrity, the U.S. Nuclear Regulatory Commission (NRC) requires the performance of periodic inservice inspections (ISIs) of SG tubes. These inspections detect, in part, flaws in the tubes resulting from interaction with the SG operating environment. ISIs may also provide a means of characterizing the nature and cause of any tube flaws so that corrective measures can be taken. Tubes with flaws that exceed the tube repair criteria specified in a plant's TSs are removed from service by plugging. The TSs provide the acceptance criteria related to the results of SG tube inspections.

The requirements for the inspection of SG tubes are intended to ensure that this portion of the reactor coolant system maintains its integrity. Tube integrity means that the tubes are capable of performing their design functions in accordance with the plant design and licensing basis. Tube integrity includes both structural and leakage integrity. Structural integrity refers to maintaining adequate margins against gross failure, rupture, and collapse of the SG tubes. Leakage integrity refers to limiting primary-to-secondary leakage during normal operation, plant transients, and postulated accidents, to within the limits of the updated final safety analysis report and the plant's TSs. These limits ensure the radiological dose consequences associated with any leakage are within acceptable limits and they limit the frequency of SG tube ruptures.

In reviewing requests of this type, the NRC staff verifies that a methodology exists that maintains the structural and leakage integrity of the tubes consistent with the plant design and licensing basis. This includes verifying that the applicable General Design Criteria (GDC) (e.g., GDC 14 and 32) contained in Appendix A of Part 50 to Title 10 of the *Code of Federal Regulations* (10 CFR) and the performance criteria in the plant TSs are satisfied. The NRC staff's evaluation also includes verifying that a methodology exists for determining the amount of primary-to-secondary leakage that may occur during design-basis accidents. The amount of leakage is limited to ensure that offsite and control room dose criteria are met. The radiological dose criteria are specified, in part, in 10 CFR Part 100, 10 CFR Section 50.67, and GDC 19 of Appendix A to 10 CFR Part 50.

The regulations in 10 CFR 50.36 establish regulatory requirements related to the content of TSs. Pursuant to 10 CFR 50.36, TS are required to include items in the following five specific categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; and (5) administrative controls.

The NRC approved a similar repair criteria for the original SGs at the Kewaunee Power Station in 1996 (ADAMS Accession No. ML020780433); for the Joseph M. Farley Plant, Unit 2, in 1996 (ADAMS Accession No. ML013170044); for the Comanche Peak Steam Electric Station, Unit 1, in 1999 (ADAMS Accession No. ML021820316); for the Watts Bar Nuclear Plant, Unit 1, in 2000 (ADAMS Accession No. ML003748725); and for the Beaver Valley Power Station, Unit 2, in 2011 (ADAMS Accession No. ML110350162). In each case, plant-specific repair criteria were determined.

3.0 TECHNICAL EVALUATION

3.1 Background

There are four Westinghouse Model D3 recirculating pre-heater type SGs at WBN Unit 2. Each SG contains 4,674 mill annealed Alloy 600 tubes that have a nominal outer diameter of 0.75 inches and a nominal wall thickness of 0.043 inches. These SGs are the same design as the original WBN Unit 1 SGs. The SGs have a vertical shell and U-tube evaporator with integral moisture separating equipment. The SG tubesheet is approximately 21 inches thick, and the tubes are expanded for the full depth of the tubesheet. The tube expansion is accomplished with a rolling process, and the joint is referred to as a hard-roll expansion.

Each tube-to-tubesheet joint consists of a tube, the tubesheet, and a tube-to-tubesheet weld, which is located at the end of the tube. Typically, the tube-to-tubesheet joints in an SG are designed as welded joints rather than friction joints. That is, the tube-to-tubesheet weld itself is designed as the pressure boundary element that transmits the entire differential pressure load from the tube to the tubesheet, with no credit taken for the friction developed between the expanded tube and tubesheet. In addition, the weld makes the joint leaktight.

Primary water stress corrosion cracking (PWSCC) of mill annealed Alloy 600 tubing has occurred. The conditions necessary for this type of cracking are primarily in two locations in the SGs. The first location is in the U-bend of the tubes in rows 1 and 2, where the residual stresses from tube bending may be high enough to lead to PWSCC. The second location is in the area where the tube is hard-roll expanded into the tubesheet, where the residual stresses from rolling can lead to PWSCC.

To provide additional margin against PWSCC, procedures were performed for reduction of residual tensile stresses at or near the inner surface of the tubes at WBN Unit 2. The rows 1 and 2 U-bends, including both tangent points, have had a thermal stress relief cycle applied by use of a resistance heater. The roller-expanded portion of the tubes within the hot-leg portion of the tubesheet, and at the transition between the expanded and non-expanded portion of the tubes at the top of the hot-leg portion of the tubesheet at WBN Unit 2, have had a mechanical stress modification by the application of the process known as rotopeening. The identical rotopeening (hot-leg) process was also applied on the original WBN Unit 1 Model D3 SGs. The tube expansion processes used on the original SGs at WBN Units 1 and 2 are identical, and the rotopeening process was implemented for both units prior to operation.

The existing inspection and repair requirements in the plant TSs do not take into account the reinforcing effect of the tubesheet on the external surface of an expanded SG tube. Nonetheless, the presence of the tubesheet constrains the tube and complements tube integrity in that region by preventing tube deformation beyond the expanded outside diameter of the tube. The resistance to both tube rupture and tube collapse is significantly enhanced by the tubesheet reinforcement. In addition, the proximity of the tubesheet to the expanded tube significantly reduces the leakage from any through-wall defect in the tube.

Based on these considerations, power reactor licensees have proposed, and the NRC has approved, alternate repair criteria (ARC) for SG tube defects located in the lower portion of the tube within the tubesheet, when these defects are a specific distance below the bottom of the roll transition (BRT) or the top of the tubesheet (TTS), whichever is lower. The F* methodology defines a distance, referred to as the F* distance, such that any type or combination of flaws below this distance (including flaws in the tube-to-tubesheet weld) are considered acceptable. That is, even if inspections below the F* distance identify flaws, the regulatory requirements pertaining to tube structural and leakage integrity would be met, provided there were no significant flaws within the F* distance. The F* distance is measured from the TTS, or the BRT, whichever is lower.

Determination of the F* distance includes non-destructive examination (NDE) uncertainty to account for uncertainty in determining the location of the flaw relative to the BRT or TTS.

The F* analysis considered the forces acting to pull the tube out of the tubesheet (i.e., from the internal pressure in the tube) and the forces acting to keep the tube in place. These latter forces are a result of friction and the forces arising from (1) the residual preload from the installation (rolling) process, (2) the differential thermal expansion between the tube and the tubesheet, and (3) internal pressure in the tube within the tubesheet. In addition, the effects of tubesheet bow, due to pressure and thermal differentials across the tubesheet, were considered, since this bow causes dilation of the tubesheet holes from the secondary face to approximately the midpoint of the tubesheet and reduces the ability of the tube to resist pullout. The amount of tubesheet bow varies as a function of radial position, with locations near the periphery experiencing less bow. The effects of tubesheet hole dilation were analyzed using the worst case hole (location) in the tubesheet. In addition, the secondary side pressure was assumed to act in the tube-to-tubesheet crevice when calculating the pressure preload.

3.2 Tennessee Valley Authority Proposal

The technical basis for the F* criterion for WBN Unit 1 was originally presented in Westinghouse report, WCAP-13084 (proprietary), "Tubesheet Region Tube Alternate Plugging (F*) Criterion for the Tennessee Valley Authority Watts Bar Units 1 and 2 Nuclear Power Plant Steam Generators," dated October 1991 and WCAP-13085 (non-proprietary). The NRC approved use of the F* ARC for WBN Unit 1 on September 8, 2000. Subsequent to the licensing of F* for the original WBN Unit 1 SGs, the H* ARC was approved for plants with hydraulically expanded tubes within the tubesheet. The licensee's current proposal to implement F* for the WBN Unit 2 SGs considers the pertinent NRC requests for additional information that that were developed during the H* licensing process to show that these requests for additional information either do not affect or are addressed by the current F* analysis for WBN Unit 2. The technical basis for the current WBN Unit 2 proposal is contained in Westinghouse Report SG-SGMP-13-15-P (proprietary), Revision 0, "Watts Bar Nuclear Plant Unit 2 F* Alternate Repair Criterion Technical Support Document," dated March 2014, and SG-SGMP-13-15-NP (non-proprietary), dated March 2015. The evaluation of the technical basis discussed in Sections 3.3. and 3.4 of this safety evaluation is based on the Westinghouse Report SG-SGMP-13-15-P, Revision 0, dated March 2014, unless otherwise noted.

Proposed TS Changes:

Section 5.7.2.12c. of the existing TSs indicates that tubes found by ISI to contain a flaw with a depth equal to or exceeding 40 percent of the nominal tube wall thickness shall be plugged. The proposed amendment adds and applies ARC to all SG tubes and defines the ARC in proposed TS 5.7.2.12c.1, which states:

> The following alternate tube repair criteria shall be applied as an alternative to the 40% depth based criteria:
>
> 1. Tubes with service-induced flaws located in the portion of the tube from the top of the tubesheet to 1.64 inches below the top of the tubesheet, or from the bottom of the roll transition to 1.64 inches below the bottom of the roll transition, whichever is lower, shall be plugged. Tubes with service-induced flaws located below this elevation do not require plugging.

Proposed addition to TS 5.7.2.12d. states that:

> The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from 1.64 inches below the bottom of the roll transition or 1.64 inches below the top of the tubesheet, whichever is lower at the tube inlet, to 1.64 inches below the bottom of the roll transition or 1.64 inches below the top of the tubesheet, whichever is lower at the tube outlet, and that may satisfy the applicable tube plugging criteria."

The following sections describe the NRC staff's evaluation of the WBN Unit 2 F* proposal in terms of maintaining SG structural and leakage integrity.

3.3 Tube Structural Integrity

The proposed amendment would permit tubes with flaws to remain in service; therefore, the licensee must demonstrate that the tubes kept in service using the F* methodology will maintain adequate structural integrity for the period between inspections. Tube rupture and pullout of a tube from the tubesheet are the two potential credible modes of structural failure considered for tubes returned to service under the F* methodology.

In order for a tube to rupture, a flaw would need to grow above the tubesheet's secondary face. If the entire flaw remains within the tubesheet, the reinforcement provided by the tubesheet will prevent tube rupture. The F* methodology proposed by the licensee for WBN Unit 2 requires the plugging of any tubes with service-induced flaws in the top portion of the tube within the tubesheet.

Therefore, any known flaws remaining in service following the inspections will be located a minimum of 1.64 inches below the TTS or the BRT, whichever is lower. Industry operating experience shows flaw growth rates within the tubesheet are typically below those necessary to propagate a flaw from 1.64 inches below the TTS to outside the tubesheet in one operating cycle (typically 18 months). Therefore, it is unlikely that any of these flaws will grow in an axial direction and extend outside the tubesheet during one operating cycle. Thus, tube burst is precluded for these flaws due to the reinforcement provided by the surrounding tubesheet.

In the event that undetected flaws are present in the F* distance, or that new flaws initiate in the F* distance during the operating cycle following an inspection, it is possible that these flaws could grow in the axial direction and extend outside the tubesheet. As a result, the NRC staff considered the conditions that would be necessary for structural failure of a tube with this type of flaw. SG tube rupture is primarily a function of flaw geometry, the differential pressure across the tube wall, and the flaw location. Axial through-wall flaws may result in a tube failing to maintain adequate margins for burst under all operating conditions. However, this would require the flaws to exceed a certain length, typically about one-half inch or longer, and have no external restraint (i.e., occur in the free span). Partially through-wall flaws would require additional length beyond the one-half inch postulated above in order to become susceptible to spontaneous rupture, based on empirical models for tube burst. Thus, these flaws would have to extend a significant distance above the tubesheet to degrade the margins of structural integrity for the affected tube (i.e., tubes with undetected flaws slightly below the TTS).

In addition, constraining a flaw at one end by the tubesheet would further elevate the burst pressure of this tube (compared to an identical flaw with no constraint). Flaw growth rates necessary for undetected or newly initiated flaws to reach a critical flaw size are unlikely to occur, given the inspections that are required to be performed. Therefore, flaws remaining in service under either of the two scenarios described above should result in the tube(s) maintaining adequate margins for tube burst.

The other postulated structural failure mode for tubes remaining in service using the F* methodology is pullout of the tube from the tubesheet due to axial loading on the tube. The differential pressure between the primary and secondary sides of an SG result in axial loads being applied to each tube, and these axial loads are then transferred to the tube-to-tubesheet joint. Axial tube loading during normal operating conditions can be significant. The peak postulated loading, however, occurs during events involving a depressurization of the secondary side of the SG, such as a feedline break or main steamline break. The presence of flaws within a tube decreases the load-bearing capability of the affected tube. If a tube becomes sufficiently degraded, these loads could lead to an axial separation of the tube.

The analysis supporting the licensee's proposed modifications to the tube inspection requirements addressed the limiting conditions necessary to maintain adequate structural integrity of the tube-to-tubesheet joint. Specifically, the tube must not experience excessive displacement relative to the tubesheet under bounding loading conditions with appropriate factors of safety considered. Safety factor criteria are derived from the American Society of Mechanical Engineers Boiler & Pressure Vessel Code, Section III.

To justify the structural integrity acceptability of any flaw or combination of flaws below the F* distance, the licensee completed an assessment using analytical calculations and laboratory experiments. This assessment included measurements of the elastic radial preload, due to the hard-roll expansion process, using tube sections rolled into simulated tubesheets (collars). Physical dimensions were measured before and after rolling the tube sections into the collars and then again after removing the collar. The amount of tube deflection was analyzed to determine the amount of preload radial stress present following the rolling process. The assessment also included calculations of the changes in radial preload during normal operating and faulted conditions due to thermal expansion, differential pressure, and tubesheet bowing. The required engagement distance, the F* length, was then calculated by equating the load-carrying ability of the tube (preload frictional forces) to the applied operating loads. These calculations included a reduction in the load-carrying ability near the ends of a severed tube.

The F* lengths (not including NDE uncertainty) calculated for normal operating and faulted SG conditions were 1.12 inches and 1.16 inches, respectively. These values were determined using a safety factor of 3 for normal operating conditions and 1.4 for faulted conditions. The most limiting of these values was used to specify the F* length of 1.16 inches in determining the required engagement length of tubing. Determination of the F* distance includes an NDE uncertainty value of 0.34 inches, which was established in Westinghouse Report WCAP-13085, dated October 1991, and remained bounding under the more recent Westinghouse Report

SG-SGMP-13-15-NP, dated March 2015. The safety factors used for normal operating and faulted conditions are consistent with the plant TSs, and the pressure differentials used bound those at WBN Unit 2; therefore, the NRC staff finds them acceptable.

Based on the above, the NRC staff has concluded that the F* repair criteria will not compromise the structural integrity of the tubes.

3.4 Tube Leakage Integrity

In assessing leakage integrity of an SG under postulated accident conditions, the leakage from all sources must be assessed. The combined leakage from all sources is limited by a plant-specific limit that is based primarily on radiological dose consequences. This limit is referred to as the accident-induced leakage limit. The licensee's approach to addressing leakage from flaws within the tubesheet region considers two regions: (1) the upper portion of the tube that is within 1.64 inches of the TTS or within 1.64 inches of the BRT, whichever is lower, and (2) the region more than 1.64 inches below the TTS, or more than 1.64 inches below the BRT, whichever is lower. In general, the licensee assumes there will be no leakage from either region. As discussed below, the NRC staff determined that the leakage from either region would not be significant.

After the F* ARC was approved for WBN Unit 1 in 2000, significant research and testing was performed to approve a similar ARC for SG tubes that were hydraulically expanded. The result of this effort was the H* ARC, which was first approved for permanent use in 2012.[1] Room temperature leakage testing of H* specimens showed that tube-in-tubesheet engagement lengths as long as 16.5 inches included measurable leakage. Prediction of the measured leak rates using fluid mechanics formulas, which included only the effects of crevice length, flow area, and fluid viscosity, showed a very good correlation with the test data, indicating that relatively low residual contact pressure between the tube and tubesheet was generated by the hydraulic expansion process. By comparison, room and elevated temperature leakage testing of roll expanded tube samples showed that roll engagement lengths as short as 0.50 inches produced a leaktight condition, at both normal operating and faulted conditions, for a portion of the specimens. The same equations used to describe the H* tests were applied to the F* leakage data and showed that crevice widths of approximately 9×10^{-6} inches would be required for the 0.50-inch F* specimens that leaked. This suggests that any leakage from F* tubes would be due to surface imperfections of the tubesheet hole and would, therefore, be exceptionally small, such that any contribution to primary-to-secondary leakage estimates from F* tubes could be neglected.

For the region below the F* distance, the licensee's evaluation considered the effects of the hard-roll expansion installation, the primary-to-secondary pressure differential, differential thermal expansion, and tubesheet bow on the interference fit between the tube and tubesheet, and compared these effects with the leakage driving force from the primary-to-secondary pressure differential. These tests indicated that at 1.64 inches or more below the TTS, or

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1. Section 1.3.3, page 1-12, of NUREG-2188, "U.S. Operating Experience with Thermally Treated Alloy 600 Steam Generator Tubes Through December 2013," dated February 29, 2016 (ADAMS Accession No. ML16061A159).

1.64 inches or more below the BRT, the contact pressure for all tubes will be higher than the highest anticipated leakage driving pressure of 2,650 pounds per square inch, corresponding to a main steamline break.

In summary, the NRC staff concludes that the proposed F* distance is acceptable and ensures that the amount of accident-induced leakage from flaws below the F* distance will be negligible, compared to the leakage rate assumed in the licensee's accident analyses. The NRC has previously approved similar F* amendments for other plants that assumed negligible accident-induced leakage from flaws below the F* distance.

3.5 Summary

The NRC staff has reviewed the license amendment proposed by the licensee and the test results and calculations that support implementation of the F* ARC. Based on review of the licensee's submittal, the NRC staff finds that the TS changes proposed for implementing the F* ARC meet the regulatory requirements in 10 CFR 50.36(c)(5), Administrative Controls; that the F* criterion will provide adequate structural and leakage integrity for tubes with flaws below the F* distance; and that tubes with flaws below the F* distance may remain in service. Therefore, the NRC staff finds the proposed TS changes to be 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. 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. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (81 FR 7844; February 16, 2016). 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 amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: Andrew Johnson

Date: September 6, 2016

September 6, 2016

Mr. Joseph W. Shea
Vice President, Nuclear Licensing
Tennessee Valley Authority
1101 Market Street, LP 3R-C
Chattanooga, TN 37402-2801

SUBJECT: WATTS BAR NUCLEAR PLANT, UNIT 2 – ISSUANCE OF AMENDMENT REGARDING REVISED STEAM GENERATOR INSPECTION SCOPE USING THE F* METHODOLOGY (CAC NO. MF7218)

Dear Mr. Shea:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 2 to Facility Operating License No. NPF-96 for the Watts Bar Nuclear Plant, Unit 2. This amendment consists of changes to the license in response to your application dated December 15, 2015, as supplemented by letters dated May 4, 2016, and June 1, 2016.

The amendment revises the Technical Specifications to allow implementation of the F* (F-star) alternate repair criterion for steam generator tubes.

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

If you have any questions regarding this letter, please contact me at (301) 415-6020 or Robert.Schaaf@nrc.gov.

Sincerely,
/RA/
Robert G. Schaaf, 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. 2 to NPF-96
2. Safety Evaluation

cc w/enclosures: Distribution via Listserv

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OFFICE	NRR/DORL/LPL2-2/PM	NRR/DORL/LPL2-2/LA	NRR/DE/ESGB/BC(A)*	NRR/DSS/STSB/BC
NAME	RSchaaf	BClayton (LRonewicz for)	PKlein	AKlein
DATE	8/18/2016	8/11/2016	6/13/2016	8/19/2016
OFFICE	OGC	NRR/DORL/LPL2-2/BC(A)	NRR/DORL/LPL2-2/PM	
NAME	VHoang	TOrf	RSchaaf	
DATE	8/31/2016	9/6/2016	9/6/2016	

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