Tennessee Valley Authority, 1101 Market Street, Chattanooga, Tennessee 37402

CNL-18-128

November 8, 2018

10 CFR 50.90

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

Watts Bar Nuclear Plant, Unit 2
Facility Operating License No. NPF-96
NRC Docket No. 50-391

Subject: **Response to Request for Additional Information Regarding Application to Revise Watts Bar Nuclear Plant Unit 2 Technical Specifications for Use of voltage-based Alternate Repair Criteria in Accordance with Generic Letter 95-05 (391-WBN2-TS-17-30) (EPID L-2018-LLA-0143)**

References:

1. TVA Letter to NRC, CNL-18-003, “Application to Revise Watts Bar Nuclear Plant Unit 2 Technical Specifications for Use of voltage-based Alternate Repair Criteria in Accordance with Generic Letter 95-05 (391-WBN2-TS-17-30),” dated May 14, 2018 (ML18138A232)

2. NRC Electronic Mail to TVA, “RE: RAIs (Final) LAR to Revise the Steam Generator Technical Specifications for Watts Bar Nuclear Plant, Unit 2,” dated October 9, 2018 (ML18282A637)

In Reference 1, Tennessee Valley Authority (TVA) submitted for Nuclear Regulatory Commission (NRC) approval, a request for an amendment to Facility Operating License No. NPF-96 for the Watts Bar Nuclear Plant (WBN) Unit 2, to revise the WBN Unit 2 Technical Specification (TS) 5.7.2.12, “Steam Generator (SG) Program,” and TS 5.9.9, “Steam Generator Tube Inspection Report,” to use the voltage-based alternate repair criteria specified in the guidelines contained in Generic Letter (GL) 95-05, "Voltage-Based Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking."

In Reference 2, the Nuclear Regulatory Commission (NRC) transmitted a request for additional information (RAI) and requested a response by November 8, 2018. Enclosure 1 to this letter provides the TVA response to the RAI.

As noted in Enclosure 1 to this letter, the responses to RAIs 3, 4, and 5 require a revision to the proposed WBN Unit 2 TS changes that were provided in Reference 1. Accordingly, Enclosure 2 to this letter provides the revised WBN Unit 2 TS pages marked up to show the proposed changes. Enclosure 3 to this letter provides the existing WBN Unit 2 TS pages retyped to show the changes incorporated.

Enclosures 2 and 3 to this letter supersede the proposed WBN Unit 2 TS pages provided in Reference 1.

The enclosures to this letter do not change the no significant hazards consideration or the environmental considerations contained in Reference 1. Additionally, in accordance with 10 CFR 50.91(b)(1), TVA is sending a copy of this letter and the enclosures to the Tennessee Department of Environment and Conservation.

Enclosure 4 contains the new regulatory commitment associated with this submittal, which supplements the commitment provided in Reference 1. Please address any questions regarding this submittal to Michael A. Brown at 423-751-3275.

I declare under penalty of perjury that the foregoing is true and correct. Executed on this 8th day of November 2018.

Respectfully,

E. K. Henderson
Director, Nuclear Regulatory Affairs

Enclosures:

1. Response to Request for Additional Information Regarding Application to Revise Watts Bar Nuclear Plant Unit 2 Technical Specifications for Use of voltage-based Alternate Repair Criteria in Accordance with Generic Letter 95-05 (391-WBN2-TS-17-30) (EPID L-2018-LLA-0143)
2. Proposed TS Changes (Markups) for WBN Unit 2
3. Proposed TS Changes (Final Typed) for WBN Unit 2

cc (see Page 3)

cc (Enclosures):

NRC Regional Administrator - Region II
NRC Senior Resident Inspector - Watts Bar Nuclear Plant
NRC Project Manager – Watts Bar Nuclear Plant
Director, Division of Radiological Health - Tennessee State Department of Environment and Conservation

Response to Request for Additional Information Regarding Application to Revise Watts Bar Nuclear Plant Unit 2 Technical Specifications for Use of voltage-based Alternate Repair Criteria in Accordance with Generic Letter 95-05 (391-WBN2-TS-17-30)
(EPID L-2018-LLA-0143)

NRC Introduction

By letter dated May 14, 2018, (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18138A232), Tennessee Valley Authority (the licensee) submitted a license amendment request (LAR) to revise the steam generator (SG) technical specifications (TSs) for Watts Bar Nuclear Plant, Unit 2. The LAR, if granted, would revise the TSs to implement an alternate repair criteria for SG tubes consistent with the guidance in Generic Letter 95-05, "Voltage-Based Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking."

The repair criteria for steam generator tubes is traditionally governed by TSs that specify a depth-based criteria. The licensee is proposing to change the requirements of TS 5.7.2.12, "Steam Generator Program" and TS 5.9.9, "Steam Generator Tube Inspection Report" such that axially-oriented outside diameter stress corrosion cracking (ODSCC) that is identified to be within the confines of the tube support plate thickness are plugged in accordance with the guidance in GL 95-05. The guidance in GL 95-05 relies on empirically derived correlations between a nondestructive inspection parameter (the bobbin coil voltage) and tube burst pressure and leak rate. After a review of the licensee's LAR, the staff identified the need for additional information to complete its evaluation.

RAI 1

In the LAR dated May 14, 2018, the licensee states that "at tube-to-TSP [tube support plate] intersections with dent signals exceeding 5.0 (bobbin) volts (Section 1.b.2 of Attachment 1 of GL 95-05), and any crack indications confirmed by rotating probe coil (RPC) will be plugged." The staff identified that the statement could be interpreted in two different ways. The staff interprets it to mean that dent signals exceeding 5.0 volts coincident with crack indications confirmed by RPC are to be plugged. Please confirm that the staff interpretation is correct.

TVA Response to RA1

The NRC understanding is correct.

RAI 2

The staff identified that the licensee's LAR relied on methodologies, analyses, and evaluations that were performed for Watts Bar, Unit 1, which had the same model SGs at the time that the LAR for Unit 1 was approved (ADAMS Accession No. ML020590277). The staff's evaluation of the Unit 1 LAR included a discussion of expansion of certain inspections of dented intersections should certain conditions be met (e.g., circumferential cracking identified at a dent with a magnitude between 2.0 and 5.0 volts, circumferential cracking identified in a hot-leg dented intersection with bobbin voltage between 1.0 and 2.0 volt and the operational assessment is challenged by structural or leakage concerns).

Please discuss the criteria and plans for expansion of inspections to ensure tube integrity for dented intersections in the event that circumferential cracking is identified.

TVA Response to RAI 2

During refueling outages, if circumferential cracking is identified at a ding/dent of magnitude between three and five volts, TVA plans on performing a +Point probe examination (or qualified technique) at hot leg dented intersections greater than or equal to two volts (volts determined by bobbin coil examination). If circumferential cracking is identified at a ding/dent of magnitude between two to three volts, TVA plans on performing a +Point probe examination (or qualified technique) at hot leg dented intersections greater than or equal to one volt (as determined by bobbin coil examination).

The above criterion will be added to the WBN Unit 2 SG repair procedures following NRC approval of the referenced letter.

Circumferential cracking at dings/dents is not expected based on the operating experience with the WBN Unit 1 original SGs. However, if circumferential cracking were to become an existing degradation mechanism, it would be expected at larger voltage dings/dents. By planning to inspect at larger voltage dings/dents in the range of two to three volts, and then expanding to the one to two volts range after identifying cracks in the two to three volts range, unnecessary additional inspection scope will be minimized. The current planned inspection scope for WBN 2 Cycle 3 and later outages is to use a combo bobbin/array probe for 100% of the tubes. If this plan is used, array data will be available for all dings/dents.

Reference

TVA Letter to NRC, CNL-18-003, "Application to Revise Watts Bar Nuclear Plant Unit 2 Technical Specifications for Use of voltage-based Alternate Repair Criteria in Accordance with Generic Letter 95-05 (391-WBN2-TS-17-30)," dated May 14, 2018 (ML18138A232)

RAI 3

(a) *Approval of the voltage-based repair criteria is based, in part, on the licensee being able to successfully demonstrate, after each inspection outage, that the conditional probability of burst and the primary-to-secondary leakage during a postulated main steam line break (MSLB) will be acceptable per the guidance in GL 95-05. The licensee's application did not discuss any reporting requirements related to the proposed alternate repair criteria. Section 6.b of Attachment 1 to GL 95-05 describes information to be provided to the NRC within 90 days following each restart. Please discuss the plans for submitting information described in Section 6.b of Attachment 1 to GL 95-05 and to incorporate a corresponding reporting requirement into the Watts Bar, Unit 2 Technical Specifications or discuss an alternative to demonstrate to the NRC staff after each inspection outage that the conditional probability of burst and the primary-to-secondary leakage during a postulated MSLB will be acceptable per the guidance in GL 95-05.*

(b) *In addition, confirm that each 90-day report will identify the database used for the GL 95-05 specified calculations.*

TVA Response to RAI 3

(a) As noted in Section 3.6 of Reference 1, the proposed changes to WBN Unit 2 TS 5.9.9 were consistent with similar reporting changes approved by the NRC for WBN Unit 1 for the SG alternate repair criteria (ARC) (Reference 2). However, TVA has revised WBN Unit 2 TS 5.9.9 to include the following reporting requirement (see Enclosures 2 and 3 to this letter):

> "A report shall be submitted within 90 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with Specification 5.7.2.12, "Steam Generator (SG) Program," when voltage-based alternate repair criteria have been applied. The report shall include information described in Section 6.b of Attachment 1 to Generic Letter 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking."

(b) TVA confirms that each 90-day report will identify the latest revision of NP-7480-L, "Steam Generator Tubing ODSCC at TSP Database for Alternate Repair Limits", used for the GL 95-05 specified calculations.

References

1 TVA Letter to NRC, CNL-18-003, "Application to Revise Watts Bar Nuclear Plant Unit 2 Technical Specifications for Use of voltage-based Alternate Repair Criteria in Accordance with Generic Letter 95-05 (391-WBN2-TS-17-30)," dated May 14, 2018 (ML18138A232)

2 NRC letter to TVA, "Watts Bar Nuclear Plant, Unit 1 - Issuance of Amendment Regarding Steam Generator Tubing Alternate Repair Criteria (ARC) (TAC No. MA8635)," dated February 26, 2002 (ML020590277)

RAI 4

The proposed TS 5.7.2.12.b.2 states, in part:

> *"Leakage is not to exceed 1 gpm for the faulted SG loop and 150 gallons per day (gpd) for each unfaulted SG. For the specific types of degradation at specific locations as described in TS 5.7.2.12.c.2 of the Steam Generator Program, the leakage is not to exceed 4 gpm or the faulted SG loop and 150 gallons per day (gpd) for each unfaulted SG."*

The staff identified that the statements above seem somewhat contradictory regarding the leakage in the faulted SG loop. Furthermore, the TS can be interpreted such that the sum of leakage limits from each sentence for the unfaulted SGs (i.e., 300 gpd) is considered the acceptable limit.

After a review of the licensee's request, the staff interpretation of the proposed TS is:

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

NRC staff asked that the licensee confirm that the staff's interpretation is correct. In addition, if this is the case, modify the proposed TS wording, as appropriate.

TVA Response to RAI 4

The NRC's understanding of the proposed WBN Unit 2 TS 5.7.2.12.b.2 is correct. For clarity, the proposed WBN Unit 2 TS 5.7.2.12.b.2 has been modified as follows (see Enclosures 2 and 3 to this letter):

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

RAI 5

The proposed TS 5.7.2.12.c.2.d states, in part:

> *"…the list of tubes identified for exclusion for Unit 1 are the same as for Unit 2."*

The NRC staff identified that the tubes specified in the LAR for exclusion for Unit 1 were for the original steam generators which have since been replaced. NRC staff ask that the licensee provide more detail by possibly explicitly specifying "original steam generators" after "Unit 1" in order to provide a clear distinction in terms of the steam generators specified in TS 5.7.2.12.c.2.d.

TVA Response to RAI 5

The proposed WBN Unit 2 TS 5.7.2.12.c.2.d has been modified as NRC recommended (see Enclosures 2 and 3 to this letter).

Enclosure 2

Proposed TS Changes (Markups) for WBN Unit 2

5.7.2.12 Steam Generator (SG) Program (continued)

2. Accident induced leakage performance criterion: The primary-to-secondary accident induced leakage rate for any design basis accident, other than an SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage for all degradation mechanisms is not to exceed 150 gpd for each unfaulted SG. Leakage for all degradation mechanisms, excluding that described in Specification 5.7.2.12.c.2, is not to exceed 1 gpm in the faulted SG. Leakage for degradation mechanisms described in Specification 5.7.2.12.c.2 is not to exceed 4 gpm for the faulted SG. ~~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 plugging shall be applied as an alternative to the 40% depth based criteria:

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

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

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

(continued)

the plugging or repair limit is described below:

a) Steam generator tubes, with degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plates and FDB with bobbin voltages less than or equal to 1.0 volt will be allowed to remain in service.

b) Steam generator tubes, with degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plates and FDB with a bobbin voltage greater than 1.0 volt will be plugged or repaired, except as noted in Specification 5.7.2.12.c.2.c below.

c) Steam generator tubes, with indications of potential degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plates and FDB with a bobbin voltage greater than 1.0 volts but less than or equal to the upper voltage repair limit (calculated according to the methodology in GL 95-05 as supplemented) may remain in service if a rotating pancake coil or acceptable alternative inspection does not detect degradation.

d) Certain intersections as identified in Attachment 2 of WAT-D-10709 ("Tennessee Valley Authority, Watts Bar Nuclear Power Plant Unit 1, Application for Implementation of Voltage Based Repair Criteria, Westinghouse Steam Generator Tubes Affected by ODSCC at TSPs," Revision 0, January 12, 2000) will be excluded from application of the voltage based repair criteria as it is determined that these intersection may collapse or deform following a postulated LOCA + SSE event. As noted in Section 4 of SG-SGMP-13-16-NP, “Watts Bar Nuclear Plant Unit 2 Applicability of GL 95-05 Voltage Based Alternate Repair Criteria,” the list of tubes identified for exclusion for the Unit 1 original steam generators are the same as for Unit 2.

e) Steam generator tubes, with indications of potential degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plates and FDB with a bobbin voltage greater than the upper voltage repair limit (calculated according to the methodology in GL 95-05 as supplemented) will be plugged or repaired.

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If an unscheduled mid-cycle inspection is performed, the following mid-cycle repair limits apply instead of the limits specified in Specifications 5.7.2.12.c.2.a through 5.7.2.12.c.2.d.

The mid-cycle repair limits are determined from the following equations:

$$V_{MURL} = \frac{V_{SL}}{1.0 + NDE + Gr[(CL-\Delta t)/CL]}$$

$$V_{MLRL} = V_{MURL} - (V_{URL} - V_{LRL})[(CL-\Delta t)/CL]$$

where:

V_{MURL} =mid-cycle upper voltage repair limit based on time into cycle

V_{SL} = structural limit voltage

NDE = 95-percent cumulative probability allowance for nondestructive examination uncertainty (i.e., a value of 20 percent has been approved by NRC). The NDE is the value provided by the in GL 95-05 as supplemented.

Gr = average growth rate per cycle length

CL = cycle length (the time between two scheduled steam generator inspections)

V_{URL} = upper voltage repair limit (Note 1)

V_{LRL} = lower voltage repair limit

V_{MLRL} = mid-cycle lower voltage repair limit based on V_{MURL} and time into cycle

Δt =length of time since last scheduled inspection during which V_{URL} and V_{LRL} were implemented

Implementation of these mid-cycle repair limits should follow the same approach as in Specifications 5.7.2.12.c.2.a through 5.7.2.12.c.2.d.

Note 1: The upper voltage repair limit is calculated according to the methodology in GL 95-05 as supplemented. V_{URL} will differ at the tube support plates and flow distribution baffle.

(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.

4. Indications left in service as a result of application of the tube support plate voltage-based repair criteria (Specification 5.7.2.12.c.2) shall be inspected by bobbin coil probe during all future refueling outages.

   Implementation of the steam generator tube-to-tube support plate repair criteria requires a 100 percent bobbin coil inspection for hot-leg and cold-leg tube support plate intersections down to the lowest cold-leg tube support plate (including the FDB) with known outside diameter stress corrosion cracking (ODSCC) indications. The determination of the lowest cold-leg tube support plate intersections having ODSCC indications shall be based on the performance of at least a 20-percent random sampling of tubes inspected over their full length.

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

(continued)

5.9 Reporting Requirements (continued)

5.9.9 Steam Generator Tube Inspection Report (continued)

For implementation of the voltage based repair criteria, in accordance with GL 95-05, to tube support plate (and flow distribution baffle) intersections, notify the NRC prior to returning the steam generators to service should any of the following conditions arise:

1. If estimated leakage based on the projected end-of cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds the leakage limit (determined from the licensing basis dose calculation for the postulated main steam line break) for the next operating cycle.

2. If circumferential crack-like indications are detected at the tube support plate intersection and flow distribution baffles.

3. If indications are identified that extend beyond the confines of the tube support plate and flow distribution baffles.

4. If indications are identified at the tube support plate elevations and flow distribution baffles that are attributable to primary water stress corrosion cracking.

5. If the calculated conditional burst probability based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds 1×10^{-2} , notify the NRC and provide an assessment of the safety significance of the occurrence.

A report shall be submitted within 90 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with Specification 5.7.2.12, “Steam Generator (SG) Program, when voltage-based alternate repair criteria have been applied. The report shall include information described in Section 6.b of Attachment 1 to Generic Letter 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking.

5.10 Record Retention

(removed from Technical Specifications)

Enclosure 3

Proposed TS Changes (Final Typed) for WBN Unit 2

5.7.2.12 Steam Generator (SG) Program (continued)

2. Accident induced leakage performance criterion: The primary-to-secondary accident induced leakage rate for any design basis accident, other than an SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage for all degradation mechanisms is not to exceed 150 gpd for each unfaulted SG. Leakage for all degradation mechanisms, excluding that described in Specification 5.7.2.12.c.2, is not to exceed 1 gpm in the faulted SG. Leakage for degradation mechanisms described in Specification 5.7.2.12.c.2 is not to exceed 4 gpm for the faulted SG.

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

c. Provisions for SG tube plugging 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 plugging shall be applied as an alternative to the 40% depth based criteria:

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

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

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

(continued)

the plugging or repair limit is described below:

a) Steam generator tubes, with degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plates and FDB with bobbin voltages less than or equal to 1.0 volt will be allowed to remain in service.

b) Steam generator tubes, with degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plates and FDB with a bobbin voltage greater than 1.0 volt will be plugged or repaired, except as noted in Specification 5.7.2.12.c.2.c below.

c) Steam generator tubes, with indications of potential degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plates and FDB with a bobbin voltage greater than 1.0 volts but less than or equal to the upper voltage repair limit (calculated according to the methodology in GL 95-05 as supplemented) may remain in service if a rotating pancake coil or acceptable alternative inspection does not detect degradation.

d) Certain intersections as identified in Attachment 2 of WAT-D-10709 ("Tennessee Valley Authority, Watts Bar Nuclear Power Plant Unit 1, Application for Implementation of Voltage Based Repair Criteria, Westinghouse Steam Generator Tubes Affected by ODSCC at TSPs," Revision 0, January 12, 2000) will be excluded from application of the voltage based repair criteria as it is determined that these intersection may collapse or deform following a postulated LOCA + SSE event. As noted in Section 4 of SG-SGMP-13-16-NP, “Watts Bar Nuclear Plant Unit 2 Applicability of GL 95-05 Voltage Based Alternate Repair Criteria,” the list of tubes identified for exclusion for the Unit 1 original steam generators are the same as for Unit 2.

e) Steam generator tubes, with indications of potential degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plates and FDB with a bobbin voltage greater than the upper voltage repair limit (calculated according to the methodology in GL 95-05 as supplemented) will be plugged or repaired.

(continued)

If an unscheduled mid-cycle inspection is performed, the following mid-cycle repair limits apply instead of the limits specified in Specifications 5.7.2.12.c.2.a through 5.7.2.12.c.2.d.

The mid-cycle repair limits are determined from the following equations:

$$V_{MURL} = \frac{V_{SL}}{1.0 + NDE + Gr[(CL-\Delta t)/CL]}$$

$$V_{MLRL} = V_{MURL} - (V_{URL} - V_{LRL})[(CL-\Delta t)/CL]$$

where:

V_{MURL} =mid-cycle upper voltage repair limit based on time into cycle

V_{SL} = structural limit voltage

NDE = 95-percent cumulative probability allowance for nondestructive examination uncertainty (i.e., a value of 20 percent has been approved by NRC). The NDE is the value provided by the in GL 95-05 as supplemented.

Gr = average growth rate per cycle length

CL = cycle length (the time between two scheduled steam generator inspections)

V_{URL} = upper voltage repair limit (Note 1)

V_{LRL} = lower voltage repair limit

V_{MLRL} = mid-cycle lower voltage repair limit based on V_{MURL} and time into cycle

Δt =length of time since last scheduled inspection during which V_{URL} and V_{LRL} were implemented

Implementation of these mid-cycle repair limits should follow the same approach as in Specifications 5.7.2.12.c.2.a through 5.7.2.12.c.2.d.

Note 1: The upper voltage repair limit is calculated according to the methodology in GL 95-05 as supplemented. V_{URL} will differ at the tube support plates and flow distribution baffle.

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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.

4. Indications left in service as a result of application of the tube support plate voltage-based repair criteria (Specification 5.7.2.12.c.2) shall be inspected by bobbin coil probe during all future refueling outages.

   Implementation of the steam generator tube-to-tube support plate repair criteria requires a 100 percent bobbin coil inspection for hot-leg and cold-leg tube support plate intersections down to the lowest cold-leg tube support plate (including the FDB) with known outside diameter stress corrosion cracking (ODSCC) indications. The determination of the lowest cold-leg tube support plate intersections having ODSCC indications shall be based on the performance of at least a 20-percent random sampling of tubes inspected over their full length.

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

(continued)

5.9 Reporting Requirements (continued)

5.9.9 Steam Generator Tube Inspection Report (continued)

For implementation of the voltage based repair criteria, in accordance with GL 95-05, to tube support plate (and flow distribution baffle) intersections, notify the NRC prior to returning the steam generators to service should any of the following conditions arise:

1. If estimated leakage based on the projected end-of cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds the leakage limit (determined from the licensing basis dose calculation for the postulated main steam line break) for the next operating cycle.

2. If circumferential crack-like indications are detected at the tube support plate intersection and flow distribution baffles.

3. If indications are identified that extend beyond the confines of the tube support plate and flow distribution baffles.

4. If indications are identified at the tube support plate elevations and flow distribution baffles that are attributable to primary water stress corrosion cracking.

5. If the calculated conditional burst probability based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds 1×10^{-2} , notify the NRC and provide an assessment of the safety significance of the occurrence.

A report shall be submitted within 90 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with Specification 5.7.2.12, “Steam Generator (SG) Program, when voltage-based alternate repair criteria have been applied. The report shall include information described in Section 6.b of Attachment 1 to Generic Letter 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking.

5.10 Record Retention

(removed from Technical Specifications)

Commitment	Due Date/Event
The following criterion will added to the WBN Unit 2 Steam Generator Repair Procedures following NRC approval of the voltage-based ARC at WBN Unit 2: • During refueling outages, if circumferential cracking is identified at a ding/dent of magnitude between three and five volts, TVA plans on performing a +Point probe examination (or qualified technique) at hot leg dented intersections greater than or equal to two volts (volts determined by bobbin coil examination). If circumferential cracking is identified at a ding/dent of magnitude between two to three volts, TVA plans on performing a +Point probe examination (or qualified technique) at hot leg dented intersections greater than or equal to one volt (as determined by bobbin coil examination).	Within 60 days following NRC approval of License Amendment Request 391-WBN2-TS-17-30 (TVA letter CNL-18-003)