TENNESSEE VALLEY AUTHORITY

1101 Market Street, Chattanooga, Tennessee 37402

CNL-21-013

March 11, 2021

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: **Application to Modify Watts Bar Nuclear Plant, Unit 2 Technical Specifications Steam Generator Inspection/Repair Program Provisions and Unit 2 Facility Operating License Condition 2.C.(4) (WBN-TS-20-06)**

In accordance with the provisions of Title 10 of the *Code of Federal Regulations* (10 CFR) 50.90, "Application for amendment of license, construction permit, or early site permit," Tennessee Valley Authority (TVA) is submitting a request for an amendment to Facility Operating License (FOL) No. NPF-96 for the Watts Bar Nuclear Plant (WBN), Unit 2.

Replacement of the WBN Unit 2 Westinghouse Model D3 (Alloy 600) original steam generators (OSGs) with Westinghouse Model 68AXP (Alloy 690) replacement steam generators (RSGs) requires deletion of WBN Unit 2 Technical Specifications (TS) requirements for SG tube inspection/repair methodologies that will no longer apply following installation of the RSGs. These TS requirements include: 1) the F* SG tube inspection methodology, 2) the voltage-based alternate repair criteria (ARC) SG tube inspection methodology, and 3) the provision allowing use of SG tube sleeving as a SG tube repair methodology. This license amendment request (LAR) proposes to delete the associated WBN Unit 2 TS requirements that no longer apply following installation of the RSGs. This LAR also revises WBN Unit 2 TS 5.7.2.12.d.2, "Steam Generator (SG) Program," to reflect the TS changes in Nuclear Regulatory Commission (NRC)-approved Technical Specification Task Force (TSTF)-510, Revision 2, "Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection," for Alloy 690 thermally treated tubing. The current requirements in WBN Unit 2 TS 5.7.2.12.d.2 are based on the TSTF-510 TS requirements for Alloy 600 mill annealed tubing. Additionally, the LAR revises WBN Unit 2 FOL Condition 2.C.(4) to delete the reference to PAD4TCD, which no longer applies following installation of the RSGs.

The enclosure to this submittal provides a description and technical evaluation of the proposed change, a regulatory evaluation, and a discussion of environmental considerations.

Attachment 1 to the enclosure provides the existing WBN Unit 2 TS pages marked up to show the proposed change. Attachment 2 to the enclosure provides the existing WBN Unit 2 TS pages retyped to show the proposed change. Attachment 3 to the enclosure provides the existing WBN Unit 2 TS Bases pages marked up to show the proposed change. Changes to the existing TS Bases are provided for information only and will be implemented under the TS Bases Control Program. Attachment 4 to the enclosure provides the existing WBN Unit 2 FOL Condition 2.C.(4) marked up to show the proposed change. Attachment 5 to the enclosure provides the existing WBN Unit 2 FOL Condition 2.C.(4) retyped to show the proposed change.

TVA has determined that there are no significant hazard considerations associated with the proposed change and that the change qualifies for a categorical exclusion from environmental review pursuant to the provisions of 10 CFR 51.22(c)(9). In accordance with 10 CFR 50.91, “Notice for Public Comment; State Consultation,” TVA is sending a copy of this letter and the enclosure to the Tennessee Department of Environment and Conservation.

The WBN Unit 2 RSGs are scheduled to be installed during the WBN Unit 2 Cycle 4 refueling outage (U2R4) scheduled for spring 2022. To support the schedule for the RSG project, TVA requests NRC approval of the proposed license amendment within one year from the date of this submittal with implementation prior to entering Mode 4 during restart following the U2R4 refueling outage. The milestone of Mode 4 supports the enclosed proposed change to WBN Unit 2 TS 3.4.17, “Steam Generator (SG) Tube Integrity,” which is applicable during Modes 1, 2, 3, and 4.

There are no new regulatory commitments associated with this submittal. Please address any questions regarding this request to Kimberly D. Hulvey, Senior Manager, Fleet Licensing, at (423) 751-3275.

I declare under penalty of perjury that the foregoing is true and correct. Executed on this 11th day of March 2021.

Respectfully,

James T. Polickoski
Director, Nuclear Regulatory Affairs

Enclosure:
Evaluation of Proposed Change

cc: (Enclosure):

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

**Enclosure**

**Evaluation of Proposed Change**

**Application to Modify Watts Bar Nuclear Plant (WBN) Unit 2 Technical Specifications Steam Generator Inspection/Repair Program Provisions and Unit 2 Facility Operating License (License Condition 2.C.(4) (WBN-TS-20-06)**

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## 1.0 SUMMARY DESCRIPTION

In accordance with the provisions of Title 10 of the *Code of Federal Regulations* (10 CFR) 50.90, “Application for amendment of license, construction permit, or early site permit,” Tennessee Valley Authority (TVA) is requesting a license amendment to Facility Operating License (FOL) No. NPF-96 for the Watts Bar Nuclear Plant (WBN) Unit 2.

Replacement of the WBN Unit 2 Westinghouse Model D3 (Alloy 600) original steam generators (OSGs) with Westinghouse Model 68AXP (Alloy 690) replacement steam generators (RSGs) requires removal of the WBN Unit 2 Technical Specification (TS) requirements for SG tube inspection/repair methodologies that will no longer apply following installation of the RSGs. These TS requirements include: 1) the F* SG tube inspection methodology (Reference 1), 2) the voltage-based alternate repair criteria (ARC) SG tube inspection methodology (Reference 2), and 3) the provision allowing use of SG tube sleeving as a SG tube repair methodology (Reference 3). This license amendment request (LAR) proposes to delete the associated WBN Unit 2 TS requirements that no longer apply following installation of the RSGs. This LAR also revises WBN Unit 2 TS 5.7.2.12.d.2, “Steam Generator (SG) Program,” to reflect the TS changes in Nuclear Regulatory Commission (NRC)-approved Technical Specification Task Force (TSTF)-510, Revision 2, "Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection" for Alloy 690 thermally treated tubing (References 4, 5, and 6). The current requirements in WBN Unit 2 TS 5.7.2.12.d.2 are based on the TSTF-510 TS requirements for Alloy 600 mill annealed tubing. Additionally, the LAR revises WBN Unit 2 FOL Condition 2.C.(4) to delete the reference to PAD4TCD, which no longer applies following installation of the RSGs (Reference 7).

## 2.0 DETAILED DESCRIPTION

### 2.1 PROPOSED CHANGE

The proposed amendment revises WBN Unit 2 TS 3.4.17, “Steam Generator (SG) Tube Integrity,” TS 5.7.2.12, and TS 5.9.9, “Steam Generator Tube Inspection Report” to remove requirements related to the F* methodology, voltage-based ARC, and SG tube sleeving, that will no longer apply following installation of the RSGs. Furthermore, WBN Unit 2 TS 5.7.2.12.d.2 is revised to reflect the TSTF-510, Revision 2, TS requirements for Alloy 690 thermally treated tubing that will apply to the RSGs. The current requirements in WBN Unit 2 TS 5.7.2.12.d.2 are based on the TSTF-510 TS requirements for Alloy 600 mill annealed tubing.

Attachment 1 to this enclosure provides the existing WBN Unit 2 TS pages marked up to show the proposed change. Attachment 2 to this enclosure provides the existing WBN Unit 2 TS pages retyped to show the proposed change. Attachment 3 to this enclosure provides the existing WBN Unit 2 TS Bases pages marked up to show the proposed change. Changes to the existing TS Bases are provided for information only and will be implemented under the TS Bases Control Program.

Section 2.C.(4) of the WBN Unit 2 FOL currently states:

(4) PAD4TCD may be used to establish core operating limits until the WBN Unit 2 steam generators are replaced with steam generators equivalent to the existing steam generators at WBN Unit 1. FULL SPECTRUM LOCA Methodology shall be implemented when the WBN Unit 2 steam generators are replaced with steam generators equivalent to the existing steam generators at WBN Unit 1.

Accordingly, the proposed license amendment deletes the first sentence of this license condition because PAD4TCD no longer applies following installation of the RSGs.

Attachment 4 to this enclosure provides the existing WBN Unit 2 FOL Condition 2.C.(4) marked up to show the proposed change. Attachment 5 to this enclosure provides the existing WBN Unit 2 FOL Condition 2.C.(4) retyped to show the proposed change.

### 2.2 CONDITION INTENDED TO RESOLVE

The existing WBN Unit 2 Alloy 600 OSGs are scheduled to be replaced with Alloy 690 RSGs, as was similarly done for WBN Unit 1. Therefore, as described in Section 1.0 to this enclosure, those WBN Unit 2 TS and FOL requirements that no longer apply following installation of the RSGs need to be deleted and, as applicable, replaced with verbiage consistent with TSTF-510, Revision 2. Furthermore, the reference to PAD4TCD in FOL Condition 2.C.(4) no longer applies following installation of the RSGs.

## 3.0 TECHNICAL EVALUATION

### 3.1 SYSTEM DESCRIPTION

The WBN Unit 2 RSGs are essentially the same as the WBN Unit 1 RSGs described in Section 3.1 of Reference 8.

### 3.2 TECHNICAL ANALYSIS

As noted in References 1, 2, and 3, the F* methodology, voltage-based ARC, and SG tube sleeving are valid inspection and repair methodologies for SG tubes of Alloy 600 material. Because the WBN Unit 2 OSGs with Alloy 600 SG tubing will be replaced with RSGs that contain Alloy 690 thermally treated tubing the same as WBN Unit 1, the F* methodology, voltage-based ARC, and SG tube sleeving inspection/repair methodologies will no longer apply following installation of the RSGs in WBN Unit 2. As noted in Section 3.1 of Reference 9, which removed similar requirements from the WBN Unit 1 TS following installation of the WBN Unit 1 RSGs, “Removal of the alternate repair criteria and methods is acceptable because the licensee will be required to plug tubes that exceed the standard 40-percent depth-based repair criteria and plugging is an acceptable method for removing tubes from service.”

Furthermore, the current SG inspection interval criteria defined in WBN Unit 2 TS 5.7.2.12.d.2 also reflect the TS requirements for Alloy 600 SG mill annealed tubing as defined in TSTF-510, Revision 2. Accordingly, WBN Unit 2 TS 3.4.17, 5.7.2.12, and 5.9.9 are being revised to delete requirements related to the F* methodology, voltage-based ARC, and SG tube sleeving inspection/repair methodologies, and replace these elements with inspection and plugging criteria approved for Alloy 690 thermally treated tubing that apply for the WBN 2 RSGs in accordance with TSTF-510, Revision 2. Furthermore, WBN Unit 2 TS 5.7.2.12.d.2 is being revised to reflect the SG inspection interval criteria for Alloy 690 thermally treated tubing that is defined in TSTF-510, Revision 2.

Additionally, the reference to PAD4TCD in Section 2.C.(4) of the WBN Unit 2 FOL requires deletion because PAD4TCD no longer applies following installation of the RSGs. In Reference 10, TVA submitted a LAR to allow the continued use of PAD4TCD to establish core operating limits until the installation of the RSGs, which was approved by the NRC in Reference 7. In Reference 11, NRC approved a revision to Section 2.C.(4) of the WBN Unit 2 FOL to reflect the implementation of the FULL SPECTRUM™ Loss-of-Coolant Accident (FSLOCA™) Evaluation Model (EM) methodology, which will be implemented when the WBN Unit 2 SGs are replaced. As noted in Section 3.3.6 of Reference 11, PAD5 fuel performance data is utilized in the WBN Units 1 and 2 analysis with the FSLOCA EM. Therefore, the proposed change to Section 2.C.(4) of the WBN Unit 2 FOL supports the TVA plan for transitioning to PAD5, a model, which has been approved for use by the NRC.[1]

## 4.0 REGULATORY EVALUATION

### 4.1 APPLICABLE REGULATORY REQUIREMENTS AND CRITERIA

WBN 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 WBN construction permit was issued in January 1973. The UFSAR, however, addresses the General Design Criteria (GDC) published as Appendix A to 10 CFR 50 in July 1971. Conformance with the GDCs is described in Section 3.1.2 of the UFSAR.
Each criterion listed below is followed by a discussion of the design features and procedures that meet the intent of the criteria. Any exception to the 1971 GDC resulting from the earlier commitments is identified in the discussion of the corresponding criterion.

> *Criterion 14-Reactor coolant pressure boundary.* The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, or rapidly propagating failure, and of gross rupture.

Compliance with GDC 14 is described in Section 3.1.2.2 of the WBN UFSAR.

> *Criterion 15-Reactor coolant system design.* The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

Compliance with GDC 15 is described in Section 3.1.2.2 of the WBN UFSAR.

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[1] WCAP-17642-NP-A, Revision 1, “Westinghouse Performance Analysis and Design Model (PAD5),” November 2017 (ML17338A396 and ML17334A826)

> *Criterion 16-Containment design.* Reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

Compliance with GDC 16 is described in Section 3.1.2.2 of the WBN UFSAR.

> *Criterion 30-Quality of reactor coolant pressure boundary.* Components, which are part of the reactor coolant pressure boundary, shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

Compliance with GDC 30 is described in Section 3.1.2.4 of the WBN UFSAR.

> *Criterion 31-Fracture prevention of reactor coolant pressure boundary.* The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws.

Compliance with GDC 31 is described in Section 3.1.2.4 of the WBN UFSAR.

> *Criterion 32-Inspection of reactor coolant pressure boundary.* Components, which are part of the reactor coolant pressure boundary, shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel.

Compliance with GDC 32 is described in Section 3.1.2.4 of the WBN UFSAR.

## 4.2 PRECEDENT

The proposed change is similar to a previous license amendment for WBN Unit 1 (Reference 9), which also deleted TS requirements that no longer applied following installation of the RSG (see Section 3.1 of Reference 9).

## 4.3 NO SIGNIFICANT HAZARDS CONSIDERATION

Tennessee Valley Authority (TVA) plans to replace the Watts Bar Nuclear Plant (WBN) Unit 2 Westinghouse Model D3 (Alloy 600) original steam generators (OSGs) with Westinghouse Model 68AXP (Alloy 690) replacement steam generators (RSGs). Accordingly, TVA proposes to revise the WBN Unit 2 Technical Specifications (TS) to delete TS SG tube inspection/repair methodologies that will no longer apply following installation of the RSGs. These TS requirements include: 1) the F* SG tube inspection methodology, 2) the voltage-based alternate repair criteria (ARC) SG tube inspection methodology, and 3) the provision allowing use of SG tube sleeving as a SG tube repair methodology. This license amendment request (LAR) also revises WBN Unit 2 TS 5.7.2.12.d.2, “Steam Generator (SG) Program,” to reflect the TS changes in Nuclear Regulatory Commission (NRC)-approved Technical Specification Task Force (TSTF)-510, Revision 2, "Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection," for Alloy 690 thermally treated tubing. The current requirements in WBN Unit 2 TS 5.7.2.12.d.2 are based on the TSTF-510 TS requirements for Alloy 600 mill annealed tubing. Additionally, the LAR revises WBN Unit 2 Facility Operating License (FOL) Condition 2.C.(4) to delete the reference to PAD4TCD, which no longer applies following installation of the RSGs.

TVA has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, “Issuance of amendment,” as discussed below:

1. *Does the proposed amendment involve a significant increase in the probability or consequence of an accident previously evaluated?*

   **Response: No**
   The proposed amendment deletes WBN Unit 2 TS and FOL requirements that are no longer applicable following installation of the WBN Unit 2 RSGs and reflect TS requirements consistent with NRC approved methodologies for SGs containing Alloy 690 thermally treated tubes. The proposed SG tube inspection frequency and sample selection criteria will continue to ensure that the SG tubes are inspected such that the probability of a SGTR is not increased. The consequences of a SGTR continue to be bounded by the conservative assumptions in the design basis accident analysis. The proposed change will not cause the consequences of a SGTR to exceed those assumptions.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. *Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?*

   **Response: No.**

   The proposed amendment deletes WBN Unit 2 TS and FOL requirements that are no longer applicable following installation of the WBN Unit 2 RSGs and reflect TS requirements consistent with NRC approved methodologies for SGs containing Alloy 690 thermally treated tubes. The proposed amendment does not affect the method of operation of the SGs, modify the reactor coolant pressure boundary, or the primary or secondary coolant chemistry controls. In addition, the proposed amendment does not impact any other plant system or component. The proposed amendment modifies existing SG inspection requirements based on the RSG design and the properties and experience associated with their improved materials. The revised inspection requirements will result in the same outcome that SG tube integrity will continue to be maintained.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. *Does the proposed amendment involve a significant reduction in a margin of safety?*

   **Response: No.**

   The proposed amendment deletes WBN Unit 2 TS and FOL requirements that are no longer applicable following installation of the WBN Unit 2 RSGs and reflect TS requirements consistent with NRC approved methodologies for SGs containing Alloy 690 thermally treated tubes. The SG tubes in pressurized water reactors are an integral part of the reactor coolant pressure boundary and, as such, are relied upon to maintain the primary system's pressure and inventory. As part of the reactor coolant pressure boundary, the SG tubes are unique in that they are also relied upon as a heat transfer surface between the primary and secondary systems such that residual heat can be removed from the primary system. In addition, the SG tubes isolate the radioactive fission products in the primary coolant from the secondary system. In summary, the safety function of a SG is maintained by ensuring the integrity of its tubes. SG tube integrity is a function of the design, environment, and the physical condition of the tube. The proposed change to the SG inspection program does not affect tube design or operating environment. The repair criteria that are being removed are specific to the existing SGs and with the commensurate criteria being added for the RSGs. For the above reasons, the margin of safety is not changed.

   Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, TVA concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92 (c), and accordingly, a finding of “no significant hazards consideration” is justified.

### 4.4 Conclusion

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) 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.

## 5.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

## 6.0 REFERENCES

1. NRC Letter to TVA, "Watts Bar Nuclear Plant, Unit 2 – Issuance of Amendment Regarding Revised Steam Generator Inspection Scope Using the F* Methodology (CAC No. MF7218)," dated September 6, 2016 (ML16203A365)
2. NRC Letter to TVA, "Watts Bar Nuclear Plant, Unit 2 – Issuance of Amendment Regarding Application to Revise Technical Specifications for Use of Voltage-Based Alternate Repair Criteria in Accordance with Generic Letter 95-05 (EPID L-2018-LLA-0143)," dated June 3, 2019 (ML19063B721)
3. NRC Letter to TVA, "Watts Bar Nuclear Plant, Unit 2 – Issuance of Amendment No. 40 Regarding Technical Specifications for Steam Generator Tube Repair Sleeve (EPID L-2019-LLA-0209)," dated August 10, 2020 (ML20156A018)
4. Technical Specifications Task Force (TSTF) Letter No. TSTF-11-02, "Correction to TSTF-510, Revision 2, 'Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection'," dated March 1, 2011 (ML110610350)
5. Federal Register Notice: Notice of Availability of Models for Plant-Specific Adoption of TSTF-510, Revision 2, "Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection," dated October 19, 2011 (ML112101604)
6. Technical Specifications Task Force letter to NRC, TSTF-12-09, "Correction to TSTF-510-A, Revision 2, 'Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection'," dated March 28, 2012 (ML12088A082)
7. NRC Letter to TVA, "Watts Bar, Unit 2 – Issuance of Amendment Regarding Application to Revise License Condition 2.C.(4) PAD4TCD (EPID L-2018-LLA-0051)," dated March 20, 2019 (ML19046A286)
8. TVA letter to NRC, CNL-20-053, "Application to Revise Watts Bar Nuclear Plant (WBN), Unit 1 Technical Specifications for Steam Generator Tube Inspection Frequency and to Adopt TSTF-510, 'Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection,' (WBN-390-TS-20-012)," dated July 17, 2020 (ML20199M346)
9. NRC Letter to TVA, "Watts Bar Nuclear Plant, Unit 1 – Issuance of Amendment Regarding Steam Generator Tube Integrity (WBN-TS-05-10) (TAC No. MC9271)," dated November 3, 2006 (ML062910093)
10. TVA letter to NRC, CNL-18-016, "Watts Bar Nuclear Plant Unit 2 - Application to Revise License Condition 2.C(4) PAD4TCD (391-WBN-TS-18-03)," dated March 5 2018 (ML18064A192)
11. NRC letter to TVA, "Watts Bar Nuclear Plant, Units 1 and 2 - Issuance of Amendment Nos. 143 and 50 Regarding Implementation of FULL SPECTRUM™ Loss-of-Coolant Accident Analysis (LOCA) and New LOCA-Specific Tritium Producing Burnable Absorber Rod Stress Analysis Methodology (EPID L-2020-LLA-0005)," dated February 26, 2021 (ML21034A166)

Attachment 1

Proposed TS Changes (Markups) for WBN Unit 2

## 3.4 REACTOR COOLANT SYSTEM (RCS)

### 3.4.17 Steam Generator (SG) Tube Integrity

LCO 3.4.17 SG tube integrity shall be maintained

<u>AND</u>

All SG tubes satisfying the tube plugging ~~or repair~~ criteria shall be plugged ~~or repaired~~ in accordance with the Steam Generator Program.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

-----------------------------------------------------------NOTE-----------------------------------------------------------
Separate Condition entry is allowed for each SG tube.
--------------------------------------------------------------------------------------------------------------------------

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| A. One or more SG tubes satisfying the tube plugging ~~or repair~~ criteria and not plugged ~~or repaired~~ in accordance with the Steam Generator Program. | A.1 | Verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection. | 7 days |
| | <u>AND</u> | | |
| | A.2 | Plug ~~or repair~~ the affected tube(s) in accordance with the Steam Generator Program. | Prior to entering MODE 4 following the next refueling outage or SG tube inspection. |
| B. Required Action and associated Completion Time of Condition A not met.<br><br><u>OR</u><br><br>SG tube integrity not maintained. | B.1 | Be in MODE 3. | 6 hours |
| | <u>AND</u> | | |
| | B.2 | Be in MODE 5. | 36 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.4.17.1 | Verify steam generator tube integrity in accordance with the Steam Generator Program. | In accordance with the Steam Generator Program. |
| SR 3.4.17.2 | Verify that each inspected SG tube that satisfies the tube plugging ~~or repair~~ criteria is plugged ~~or repaired~~ in accordance with the Steam Generator Program. | Prior to entering MODE 4 following a SG tube inspection. |

5.7 Procedures, Programs, and Manuals (continued)

5.7.2.12 Steam Generator (SG) Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during a SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging ~~or repair~~ of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected~~,~~ or plugged, ~~or repaired~~ to confirm that the performance criteria are being met.

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

   1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cooldown), all anticipated transients included in the design specification and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

(continued)

5.7 Procedures, Programs, and Manuals

5.7.2.12 Steam Generator (SG) Program (continued)

2. Accident induced leakage performance criterion: The primary-to-secondary accident induced leakage rate for any design basis accident, other than an SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage for all degradation mechanisms is not to exceed 150 gpd for each unfaulted SG. Leakage for all degradation mechanisms~~, 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 ~~or repair~~ criteria. Tubes found by inservice inspection to contain ~~a~~ flaws ~~in a non-sleeved region~~ with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged ~~or repaired~~.

c.

~~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 or repaired.~~

~~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:~~

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

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

~~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 valve provided by the 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~~

~~$\Delta t$ = 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)~~(continued)

5.7 Procedures, Programs, and Manuals

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 t~~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~~he tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube plugging ~~or repair~~ criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. A degradation assessment shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

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

(continued)

2. After the first refueling outage following SG installation, inspect each SG at least every ~~24~~72 effective full power months or at least every third refueling outage (whichever results in more frequent inspections). In addition, the minimum number of tubes inspected at each scheduled inspection shall be the number of tubes in all SGs divided by the number of SG inspection outages scheduled in each inspection period as defined in a, b, c and d below. If a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type~~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 or repair criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated.~~ of degradation at this location and that may satisfy the applicable tube plugging criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated. The fraction of locations to be inspected for this potential type of degradation at this location at the end of the inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the inspection period after the determination that a new form of degradation could potentially be occurring at this location divided by the total number of times the SG is scheduled to be inspected in the

(continued)

5.7 Procedures, Programs, and Manuals

5.7.2.12 Steam Generator (SG) Program (continued)

inspection period. Each inspection period defined below may be extended up to 3 effective full power months to include a SG inspection outage in an inspection period and the subsequent inspection period begins at the conclusion of the included SG inspection outage.

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

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

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

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

~~(continued)~~
(continued)

5.7 Procedures, Programs, and Manuals

~~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)~~

(continued)

5.7 Procedures, Programs, and Manuals

~~5.7.2.12 Steam Generator (SG) Program (continued)~~

~~f. Provisions for SG Tube Repair Methods:~~

~~Steam generator tube repair methods shall provide the means to reestablish the RCS pressure boundary integrity of SG tubes without removing the tube from service. For the purposes of these Specifications, tube plugging is not a repair. All acceptable tube repair methods are listed below.~~

~~1. Westinghouse leak-limiting Non-Nickel Banded Alloy 800 sleeves, WCAP-15918-P, Revision 4, "Steam Generator Tube Repair for Combustion Engineering and Westinghouse Designed Plants with ¾ Inch Inconel 600 Tubes Using Leak Limiting Alloy 800 Sleeves." A Non-Nickel Banded Alloy 800 sleeve shall remain in service for no more than five fuel cycles of operation starting from the outage when the sleeve was installed.~~

~~(continued)~~

5.9 Reporting Requirements (continued)

5.9.7 DG Failures Report

If an individual diesel generator (DG) experiences four or more valid failures in the last 25 demands, these failures and any nonvalid failures experienced by that DG in that time period shall be reported within 30 days. Reports on DG failures shall include the information recommended in Regulatory Guide 1.9, Revision 3, Regulatory Position C.4, or existing Regulatory Guide 1.108 reporting requirement.

5.9.8 PAMS Report

When a Report is required by Condition B or F of LCO 3.3.3, “Post Accident Monitoring (PAM) Instrumentation,” a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.9.9 Steam Generator Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.7.2.12, Steam Generator (SG) Program. The report shall include:

a. The scope of inspections performed on each SG,

b. Degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

d. Location, orientation (if linear), and measured sizes (if available) of service induced indications,

e. Number of tubes plugged ~~or repaired~~ during the inspection outage for each degradation mechanism,

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

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

~~h.~~g. ~~Repair method utilized and the number of tubes repaired by each repair method.~~

~~(continued)~~

(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 x $10^{-2}$, notify the NRC and provide an assessment of the safety significance for 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)

Attachment 2

Proposed TS Changes (Final Typed) for WBN Unit 2

## 3.4 REACTOR COOLANT SYSTEM (RCS)

### 3.4.17 Steam Generator (SG) Tube Integrity

LCO 3.4.17 SG tube integrity shall be maintained

<u>AND</u>

All SG tubes satisfying the tube plugging criteria shall be plugged in accordance with the Steam Generator Program.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

---------------------------------------------------------NOTE---------------------------------------------------------
Separate Condition entry is allowed for each SG tube.
---------------------------------------------------------------------------------------------------------------------------

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| A. One or more SG tubes satisfying the tube plugging criteria and not plugged in accordance with the Steam Generator Program. | A.1 | Verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection. | 7 days |
| | <u>AND</u> | | |
| | A.2 | Plug the affected tube(s) in accordance with the Steam Generator Program. | Prior to entering MODE 4 following the next refueling outage or SG tube inspection. |
| B. Required Action and associated Completion Time of Condition A not met.<br><br><u>OR</u><br><br>SG tube integrity not maintained. | B.1 | Be in MODE 3. | 6 hours |
| | <u>AND</u> | | |
| | B.2 | Be in MODE 5. | 36 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.4.17.1 | Verify steam generator tube integrity in accordance with the Steam Generator Program. | In accordance with the Steam Generator Program. |
| SR 3.4.17.2 | Verify that each inspected SG tube that satisfies the tube plugging criteria is plugged in accordance with the Steam Generator Program. | Prior to entering MODE 4 following a SG tube inspection. |

5.7 Procedures, Programs, and Manuals (continued)

5.7.2.12 Steam Generator (SG) Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during a SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged, to confirm that the performance criteria are being met.

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

   1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cooldown), all anticipated transients included in the design specification and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

(continued)

5.7 Procedures, Programs, and Manuals

5.7.2.12 Steam Generator (SG) Program (continued)

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

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

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

(continued)

5.7.2.12 Steam Generator (SG) Program (continued)

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

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

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

(continued)

5.7 Procedures, Programs, and Manuals

5.7.2.12 Steam Generator (SG) Program (continued)

inspection period. Each inspection period defined below may be extended up to 3 effective full power months to include a SG inspection outage in an inspection period and the subsequent inspection period begins at the conclusion of the included SG inspection outage.

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

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

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

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

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

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

(continued)

5.9 Reporting Requirements (continued)

5.9.7 DG Failures Report

If an individual diesel generator (DG) experiences four or more valid failures in the last 25 demands, these failures and any nonvalid failures experienced by that DG in that time period shall be reported within 30 days. Reports on DG failures shall include the information recommended in Regulatory Guide 1.9, Revision 3, Regulatory Position C.4, or existing Regulatory Guide 1.108 reporting requirement.

5.9.8 PAMS Report

When a Report is required by Condition B or F of LCO 3.3.3, “Post Accident Monitoring (PAM) Instrumentation,” a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.9.9 Steam Generator Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.7.2.12, Steam Generator (SG) Program. The report shall include:

a. The scope of inspections performed on each SG,

b. Degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

d. Location, orientation (if linear), and measured sizes (if available) of service induced indications,

e. Number of tubes plugged during the inspection outage for each degradation mechanism,

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

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

(continued)

## 5.10 Record Retention

(removed from Technical Specifications)

Attachment 3

Proposed TS Bases Changes (Markups for Information Only) for WBN Unit 2

BASES (continued)

APPLICABLE SAFETY ANALYSES

The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding an SGTR is the basis for this Specification. The analysis of an SGTR event assumes a bounding primary to secondary LEAKAGE rate equal to the operational LEAKAGE rate limits in LCO 3.4.13, “RCS Operational LEAKAGE,” plus the leakage rate associated with a double-ended rupture of a single tube. The accident analysis for a SGTR assumes the contaminated secondary fluid is only briefly released to the atmosphere via safety valves and the majority is discharged to the main condenser.

The analysis for design basis accidents and transients other than an SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture). In these analyses, the steam discharge to the atmosphere is based on the total primary to secondary LEAKAGE of 150 gallons per day (gpd) per unfaulted steam generator and 1 gallon per minute (gpm) in the faulted steam generator. For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is assumed to be equal to the LCO 3.4.16, “RCS Specific Activity,” limits. For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of GDC 19 (Ref. 2), and 10 CFR 100 (Ref. 3) or the NRC approved licensing basis.

Steam generator tube integrity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requires that SG tube integrity be maintained. The LCO also requires that all SG tubes that satisfy the plugging ~~or repair~~ criteria be plugged ~~or repaired~~ in accordance with the Steam Generator Program.

During an SG inspection, any inspected tube that satisfies the Steam Generator Program plugging ~~or repair~~ criteria is ~~either plugged or repaired~~ removed from service by plugging. If a tube was determined to satisfy the plugging ~~or repair~~ criteria but was not plugged ~~or repaired~~, the tube may still have tube integrity.

In the context of this Specification, an SG tube is defined as the entire length of the tube, including the tube wall, between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. The tube-to-tubesheet weld is not considered part of the tube.

(continued)

ACTIONS (continued)

A.1 and A.2

Condition A applies if it is discovered that one or more SG tubes examined in an inservice inspection satisfy the tube plugging ~~or repair~~ criteria but were not plugged ~~or repaired~~ in accordance with the Steam Generator Program as required by SR 3.4.17.2. An evaluation of SG tube integrity of the affected tube(s) must be made. Steam generator tube integrity is based on meeting the SG performance criteria described in the Steam Generator Program. The SG plugging ~~or repair~~ criteria define limits on SG tube degradation that allow for flaw growth between inspections while still providing assurance that the SG performance criteria will continue to be met. In order to determine if an SG tube that should have been plugged ~~or repaired,~~ has tube integrity, an evaluation must be completed that demonstrates that the SG performance criteria will continue to be met until the next refueling outage or SG tube inspection. The tube integrity determination is based on the estimated condition of the tube at the time the situation is discovered and the estimated growth of the degradation prior to the next SG tube inspection. If it is determined that tube integrity is not being maintained, Condition B applies.

A Completion Time of 7 days is sufficient to complete the evaluation while minimizing the risk of plant operation with a SG tube that may not have tube integrity.

If the evaluation determines that the affected tube(s) have tube integrity, Required Action A.2 allows plant operation to continue until the next refueling outage or SG inspection provided the inspection interval continues to be supported by an operational assessment that reflects the affected tubes. However, the affected tube(s) must be plugged ~~or repaired~~ prior to entering MODE 4 following the next refueling outage or SG inspection. This Completion Time is acceptable since operation until the next inspection is supported by the operational assessment.

B.1 and B.2

If the Required Actions and associated Completion Times of Condition A are not met or if SG tube integrity is not being maintained, the reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the desired plant conditions from full power conditions in an orderly manner and without challenging plant systems.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.17.1

During shutdown periods the SGs are inspected as required by this SR and the Steam Generator Program. NEI 97-06, Steam Generator Program Guidelines (Ref. 1), and its referenced EPRI Guidelines, establish the content of the Steam Generator Program. Use of the Steam Generator Program ensures that the inspection is appropriate and consistent with accepted industry practices.

During SG inspections a condition monitoring assessment of the SG tubes is performed. The condition monitoring assessment determines the “as found” condition of the SG tubes. The purpose of the condition monitoring assessment is to ensure that the SG performance criteria have been met for the previous operating period.

The Steam Generator Program determines the scope of the inspection and the methods used to determine whether the tubes contain flaws satisfying the tube plugging ~~or repair~~ criteria. Inspection scope (i.e., which tubes or areas of tubing within the SG are to be inspected) is a function of existing and potential degradation locations. The Steam Generator Program also specifies the inspection methods to be used to find potential degradation. Inspection methods are a function of degradation morphology, nondestructive examination (NDE) technique capabilities, and inspection locations.

The Steam Generator Program defines the Frequency of SR 3.4.17.1. The Frequency is determined by the operational assessment and other limits in the SG examination guidelines (Ref. 6). The Steam Generator Program uses information on existing degradations and growth rates to determine an inspection Frequency that provides reasonable assurance that the tubing will meet the SG performance criteria at the next scheduled inspection. In addition, Specification 5.7.2.12 contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections. If crack indications are found in any SG tube, the maximum inspection interval for all affected and potentially affected SGs is restricted by Specification 5.7.2.12 until subsequent inspections support extending the inspection interval.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.17.2

During an SG inspection, any inspected tube that satisfies the Steam Generator Program plugging ~~or repair (Ref. 7)~~ criteria is removed from service by plugging~~either plugged or repaired~~. The tube plugging ~~or repair~~ criteria delineated in Specification 5.7.2.12 are intended to ensure that tubes accepted for continued service satisfy the SG performance criteria with allowance for error in the flaw size measurement and for future flaw growth. In addition, the tube plugging ~~or repair~~ criteria, in conjunction with other elements of the Steam Generator Program, ensure that the SG performance criteria will continue to be met until the next inspection of the subject tube(s). Reference 1 provides guidance for performing operational assessments to verify that the tubes remaining in service will continue to meet the SG performance criteria.

The Frequency of prior to entering MODE 4 following an SG inspection ensures that the Surveillance has been completed and all tubes meeting the plugging ~~or repair~~ criteria are plugged prior to subjecting the SG tubes to significant primary-to-secondary pressure differential.

REFERENCES

1. NEI 97-06, “Steam Generator Program Guidelines.”
2. 10 CFR 50 Appendix A, GDC 19, Control Room.
3. 10 CFR 100, Reactor Site Criteria.
4. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB.
5. Regulatory Guide 1.121, “Basis for Plugging Degraded Steam Generator Tubes,” August 1976.
6. EPRI, “Pressurized Water Reactor Steam Generator Examination Guidelines.”
7. ~~WCAP-15918-P, Revision 4. “Steam Generator Tube Repair for Combustion Engineering and Westinghouse Designed Plants with ¾ Inch Inconel 600 Tubes Using Leak Limiting Alloy 800 Sleeves.”~~

Attachment 4

Proposed Operating License Changes (Markups) for WBN Unit 2

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

(1) Maximum Power Level

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

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A as revised through Amendment No. 50, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. TVA shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) TVA shall implement permanent modifications to prevent overtopping of the embankments of the Fort Loudon Dam due to the Probable Maximum Flood by June 30, 2018.

(4) ~~PAD4TCD may be used to establish core operating limits until the WBN Unit 2 steam generators are replaced with steam generators equivalent to the existing steam generators at WBN Unit 1.~~ FULL SPECTRUM LOCA Methodology shall be implemented when the WBN Unit 2 steam generators are replaced with steam generators equivalent to the existing steam generators at WBN Unit 1.

(5) By December 31, 2019, the licensee shall report to the NRC that the actions to resolve the issues identified in Bulletin 2012-01, "Design Vulnerability in Electrical Power System," have been implemented.

(6) The licensee shall maintain in effect the provisions of the physical security plan, security personnel training and qualification plan, and safeguards contingency plan, and all amendments made pursuant to the authority of 10 CFR 50.90 and 50.54(p).

(7) TVA shall fully implement and maintain in effect all provisions of the Commission approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The TVA approved CSP was discussed in NUREG-0847, Supplement 28, as amended by changes approved by License Amendment No. 7.

(8) TVA shall implement and maintain in effect all provisions of the approved fire protection program as described in the Fire Protection Report for the facility, as described in NUREG-0847, Supplement 29, subject to the following provision:

Attachment 5

Proposed Operating License Changes (Final Typed) for WBN Unit 2

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

(1) Maximum Power Level

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

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A as revised through Amendment No. 50, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. TVA shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) TVA shall implement permanent modifications to prevent overtopping of the embankments of the Fort Loudon Dam due to the Probable Maximum Flood by June 30, 2018.

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

(5) By December 31, 2019, the licensee shall report to the NRC that the actions to resolve the issues identified in Bulletin 2012-01, "Design Vulnerability in Electrical Power System," have been implemented.

(6) The licensee shall maintain in effect the provisions of the physical security plan, security personnel training and qualification plan, and safeguards contingency plan, and all amendments made pursuant to the authority of 10 CFR 50.90 and 50.54(p).

(7) TVA shall fully implement and maintain in effect all provisions of the Commission approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The TVA approved CSP was discussed in NUREG-0847, Supplement 28, as amended by changes approved by License Amendment No. 7.

(8) TVA shall implement and maintain in effect all provisions of the approved fire protection program as described in the Fire Protection Report for the facility, as described in NUREG-0847, Supplement 29, subject to the following provision: