February 26, 2002

Mr. J. A. Scalice
Chief Nuclear Officer and
Executive Vice President
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, Tennessee 37402-2801

SUBJECT: WATTS BAR NUCLEAR PLANT, UNIT 1 - ISSUANCE OF AMENDMENT REGARDING STEAM GENERATOR TUBING ALTERNATE REPAIR CRITERIA (ARC) (TAC NO. MA8635)

Dear Mr. Scalice:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 38 to Facility Operating License No. NPF-90 for Watts Bar Nuclear Plant, Unit 1. This amendment is in response to your letter of April 10, 2000, as supplemented by letters dated September 18, 2000, August 22 and November 8, 2001, and January 15, 2002. Tennessee Valley Authority (TVA) proposed to incorporate voltage-based ARC for steam generator tubes into the Watts Bar technical specifications (TS).

The staff concludes that your proposed changes to the TS are acceptable because the proposed repair criteria follow the guidelines in Generic Letter 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected By Outside Diameter Stress Corrosion Cracking." A copy of our safety evaluation is also enclosed. Notice of issuance will be included in the Commission's biweekly *Federal Register* notice.

TVA states on page E1-5 of its submittal of April 10, 2000, that it might perform a plant-specific tube exclusion analysis at a future date based on small-break combined event loadings. The staff would find that approach acceptable in principle; however, TVA would need to seek NRC review and approval through a license amendment request.

Sincerely,

/RA/

L. Mark Padovan, Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-390

Enclosures: 1. Amendment No. 38 to NPF-90
2. Safety Evaluation

cc w/enclosures: See next page

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TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-390

WATTS BAR NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 38
License No. NPF-90

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated April 10, 2000, as supplemented by letters dated September 18, 2000, August 22 and November 8, 2001, and January 15, 2002, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-90 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 38, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. TVA shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance, and shall be implemented prior to startup following the Unit 1 Cycle 4 refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Richard P. Correia, Chief, Section 2
Project Directorate II
Division of Project Licensing Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 26, 2002

ATTACHMENT TO AMENDMENT NO. 38

FACILITY OPERATING LICENSE NO. NPF-90

DOCKET NO. 50-390

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

Remove Page	Insert Page
3.4-30	3.4-30
5.0-15	5.0-15
5.0-16	5.0-16
5.0-17	5.0-17
5.0-18	5.0-18
5.0-19	5.0-19
—	5.0-19a
—	5.0-19b
5.0-32	5.0-32 *
5.0-32a	—
5.0-32b	—
5.0-33	5.0-33 *
5.0-34	5.0-34 *
5.0-35	5.0-35
—	5.0-35a
—	B 3.4-74a
—	B 3.4-74b
B 3.4-76	B 3.4-76
B 3.4-77	B 3.4-77

* No changes were made to these pages; included only for continuity.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 38 TO FACILITY OPERATING LICENSE NO. NPF-90

TENNESSEE VALLEY AUTHORITY

WATTS BAR NUCLEAR PLANT, UNIT 1

DOCKET NO. 50-390

1.0 INTRODUCTION

Tennessee Valley Authority's (TVA's) letter of April 10, 2000, as supplemented by letters of September 18, 2000, August 22 and November 8, 2001, and January 15, 2002, submitted a request to change the technical specifications (TS) of Watts Bar Nuclear Plant, Unit 1. The request is to implement alternate repair criteria (ARC) for degraded tubes in Watts Bar steam generators. The ARC follow the guidelines set forth in U.S. Nuclear Regulatory Commission (NRC) Generic Letter (GL) 95-05, "Voltage-Based Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking." The supplemental letters provided clarifying information did not change the initial proposed no significant hazards consideration determination.

2.0 BACKGROUND

General Design Criterion 14 of Appendix A to Title10 *Code of Federal Regulations* Part 50 requires that the reactor coolant system pressure boundary be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. In addition, plant TS specify the acceptance criteria (i.e., repair limits) for degraded steam generator tubes. The traditional strategy for achieving adequate structural and leakage integrity of degraded tubes has been to establish a minimum wall thickness requirement in accordance with NRC Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR [Pressurized Water Reactor] Steam Generator Tubes." The minimum wall thickness requirement was developed assuming a uniform thinning of the tube wall. This assumed degradation mechanism is inherently conservative for certain forms of tube degradation. Conservative repair limits in the plant TS may lead to removing degraded tubes from service that may otherwise have adequate structural and leakage integrity for further service.

To reduce unnecessary conservatism in the minimum wall thickness requirement for certain degradation, the industry proposed voltage-based repair criteria for predominantly axially-oriented outside diameter stress corrosion cracking (ODSCC) confined within the thickness of the tube support plates. The staff published several conclusions regarding voltage-based repair criteria in draft NUREG-1477, "Voltage-Based Interim Plugging Criteria for Steam Generator Tubes," and in a draft GL titled "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes." The latter document was published for public comment in the *Federal Register* on August 12, 1994 (59 FR 41520). On August 3, 1995, the staff issued GL 95-05 that considered public comments on the draft GL cited above, domestic operating experience under the voltage-based repair criteria, and additional data made available from European nuclear power plants.

GL 95-05 guidance does not set depth-based limits on predominantly axially-oriented ODSCC at tube support plate locations. Instead it relies on empirically derived correlations between a nondestructive inspection parameter, the bobbin coil voltage, and tube burst pressure and leak rate. The staff recognizes that although the total tube integrity margins may be reduced following application of a voltage-based repair criteria, the guidance in GL 95-05 ensures structural and leakage integrity continue to be maintained at acceptable levels consistent with the requirements of 10 CFR Part 50 and 10 CFR Part 100. Since the voltage-based repair criteria do not require minimum tube wall thickness, tubes with through-wall cracks might remain in service. The staff included provisions for augmented steam generator tube inspections and restrictive operational leakage limits because of the increased likelihood of such flaws.

GL 95-05, in part, specifies the following for licensees:

(1) Repair criteria only applies to predominantly axially oriented ODSCC located within the bounds of the tube support plates.
(2) Perform an evaluation to confirm that the degraded steam generator tubes will retain adequate structural and leakage integrity from cycle to cycle.
(3) Adhere to specific inspection criteria to ensure consistency in methods between inspections.
(4) Periodically remove tubes from the steam generators for examination and destructive testing to verify the morphology of the degradation and provide burst and leakage data for structural and leakage integrity evaluations.
(5) Reduce the operational leakage limit in the plant TS.
(6) Implement an operational leakage monitoring program.
(7) Incorporated specific reporting requirements into the plant TS.
(8) Do not apply voltage-based repair criteria at locations where tubes with degradation could substantially deform or collapse during postulated loss-of-coolant-accident (LOCA) and safe shutdown earthquake (SSE) loading.

Watts Bar Unit 1 has four Westinghouse model D3 steam generators which use mill-annealed Alloy 600 tubing. These steam generators use carbon steel drilled-hole tube support plates and flow distribution baffle plates. The outside diameter of each tube is 3/4-inch.

3.0 EVALUATION

TVA stated that it will comply with the guidance in GL 95-05 when implementing its voltage-based alternate repair criteria. The major issues related to TVA's implementation of the alternate repair criteria are discussed below.

3.1 Tube Repair Limits

TVA's proposed repair criteria will do the following:

- Permit degraded tubes to remain in service that have indications confined to within the thickness of the tube support plates with bobbin voltages $\leq$ 1.0 volt.

- Permit degraded tubes to remain in service that have indications confined to within the thickness of the tube support plates with bobbin voltages > 1.0 volt but less than or equal to the upper voltage limit if a motorized rotating pancake coil probe or acceptable alternative inspection does not detect degradation.

- Require degraded tubes to be plugged or repaired that have indications confined to within the thickness of the tube support plates with bobbin voltages greater than the upper voltage limit.

The proposed lower voltage limit of 1.0 volt is based on using a correlation between the burst pressure and the bobbin coil voltage of pulled tube and model boiler data. This is consistent with the recommended value specified in GL 95-05 for 3/4-inch steam generator tubing. The upper voltage limit is based on the lower 95 percent prediction interval of the burst pressure versus bobbin voltage correlation, adjusted for lower bound material properties evaluated at the 95 percent confidence level. The upper voltage limit is further reduced to account for uncertainty in the nondestructive examination technique and flaw growth over the next operating cycle. The industry periodically updates the database for burst pressure and bobbin voltage when the destructive test data from pulled tubes are available; therefore, the upper voltage limit may vary as additional data are incorporated into the database. The gap between the tube and tube hole in the flow distribution baffle plates is wider than the gap in the tube support plates; therefore, the upper voltage limit for the tube indications found at the baffle plates is different than the upper repair limit for tube indications found at tube support plates

3.2 Inspection Issues

Section 3.c.3 of Attachment 1 to GL 95-05 gives guidance for probe wear. TVA proposed to use an alternative to Section 3.c.3. The alternative approach, developed through the Nuclear Energy Institute (NEI), specifies that if the probe does not satisfy the voltage variability criterion for wear of ± 15 percent limit before its replacement, all tubes which had flaw signals with voltage responses measured at 75 percent or greater of the lower repair limit must be reinspected with a bobbin probe satisfying the ± 15 percent wear standard criterion. The voltages from the reinspection should be used as the basis for tube repair. The staff completed a review of NEI's proposed alternative method and concluded that the approach is acceptable as discussed in the NRC's letter of March 18, 1996, to Alex Marion of the NEI. TVA's proposal to follow the industry approach to address probe wear is acceptable.

Industry laboratory and field studies supporting the alternative probe wear criteria showed that worn probe voltages are seldom < 75 percent of the new probe voltage for all significant voltage levels. This is discussed in the letter from Alex Marion of the NEI to the NRC dated January 23, 1996. However, in a 90-day inspection report for Byron Unit 1 dated September 9, 1996, Commonwealth Edison compared the worn probe voltage to the new probe voltage and found that the worn probe voltage was substantially < 75 percent of the new probe voltage for a few indications. Commonwealth Edison evaluated these indications and concluded that NEI's criteria to retest tubes with worn probe voltages above 75 percent of the repair limit is adequate and generally conservative due to the trend for worn probe voltages to exceed new probe voltages (as shown in the study). Comparison of the actual and projected end-of-cycle voltages did not show anything unusual attributable to the alternate probe wear criteria. The staff concludes that the aforementioned probe wear results do not indicate an immediate need to modify the industry's probe wear criteria. However, the staff will continue to monitor probe wear in the licensees' 90-day inspection reports.

With respect to the probe variability guidance in GL 95-05, TVA proposed to follow an alternative approach developed through the NEI. NEI's proposed procedures and methodology are described in the October 15, 1996, letter from the NEI to the NRC. The approach specifies that the voltage responses from the primary frequency and mix frequency channels of new probes be within ±10 percent of the nominal voltage responses when voltages are normalized to the 20 percent flaw values. The nominal voltage responses were established as the average voltages obtained from American Society of Mechanical Engineers standard drilled hole flaws for at least 10 production probes. TVA's proposal to follow the industry approach to address probe variability is acceptable.

GL 95-05 specifies that all dented tube support plate intersections with bobbin voltage > 5.0 volts should be inspected with a rotating pancake coil. If circumferential cracking or primary water stress corrosion cracking indications are detected in these dented intersections, it may be necessary to expand the rotating pancake coil sampling plan to include dented intersections with bobbin voltage < 5.0 volts. TVA stated that it will inspect all dented intersections with signals > 5.0 volts with a rotating pancake coil. For dented intersections with bobbin voltage < 5.0 volts, TVA stated that it plans to inspect hot-leg dented intersections ≥ 2.0 volts using a +point probe. If circumferential cracking is identified at a dent of magnitude between 2.0 and 5.0 volts, TVA will expand the +point inspection to hot-leg dented intersections with signals ≥ 1.0 volt. If circumferential cracking is identified in a hot-leg dented intersection with bobbin voltage between 1.0 and 2.0 volts, and the operational assessment is challenged by structural or leakage concerns, TVA will develop an expansion plan for < 1.0 volt dented intersections. TVA will notify the NRC staff of this expansion plan. The staff finds TVA's inspection plan for dented intersections acceptable because it considers the structural integrity of the tubes at the low-voltage dented intersections.

3.3 Structural and Leakage Integrity Assessments

GL 95-05 guidance for the voltage-based repair criteria focuses on maintaining tube structural integrity during the full range of normal, transient and postulated accident conditions with adequate allowance for eddy current test uncertainty and flaw growth projected to occur during the next operating cycle. RG 1.121 recommends that a margin of safety of 1.43 against tube failure under postulated accident conditions and a margin of safety of 3 against burst during normal operation be maintained for steam generator tubes. Because GL 95-05 addresses tubes affected with ODSCC confined to within the thickness of the tube support plate during normal operation, the staff concluded that the structural constraint provided by the tube support plate ensures all tubes to which the voltage-based criteria apply will retain a margin of 3 with respect to burst under normal operating conditions. For a postulated main steam line break (MSLB) accident, however, the tube support plate may displace axially during steam generator blowdown such that the ODSCC affected portion of the tubing may no longer be fully constrained by the tube support plate. Accordingly, it is appropriate to consider the ODSCC affected regions of the tubes as free standing tubes for the purpose of assessing burst integrity under postulated MSLB conditions.

In order to confirm the structural and leakage integrity of the tube until the next scheduled inspection, GL 95-05 specifies a methodology to determine the conditional burst probability and the total primary-to-secondary leak rate from an affected steam generator during a postulated MSLB event. To complete GL 95-05 prescribed assessments, TVA proposes to follow the methodology described in WCAP-14277, Revision 1, "SLB [Steam Line Break] Leak Rate and Tube Burst Probability Analysis Methods for ODSCC at TSP [tube support plate] Intersections," dated December 1996. The staff finds this to be acceptable because the methodology complies with GL 95-05 guidelines and the staff approved a similar voltage-based ARC amendment for D.C. Cook, Unit 1 (dated March 13, 1997) that referenced the methodology in WCAP-14277, Revision 1.

GL 95-05 specifies that structural and leakage integrity assessments should use the latest available database from destructive examinations of tubes removed from Westinghouse-designed steam generators. The industry and NRC established a protocol to formalize the requirements for periodically updating the industry database. TVA stated that it will follow the protocol as documented in NEI's letter of January 15, 1997, to the NRC. In addition, TVA will describe in its GL 95-05 90-day reports the database it used for GL 95-05 specified calculations. The staff finds that TVA's commitment to follow the protocol and to use the NRC-approved database to perform structural and leakage assessments are acceptable.

Guidance in GL 95-05, Section 1.b.1, states that the voltage-based repair criteria will not be applied at locations where tubes with degradation could substantially deform or collapse during postulated LOCA and SSE loading. Steam generator tubes can potentially permanently deform at Tube support plate intersections under such loading conditions. The NRC staff has previously reviewed and approved the analytical methodology to determine tube deformation or collapse in previous similar applications at Comanche Peak, Diablo Canyon and Sequoyah nuclear plants. TVA used an identical methodology at Watts Bar to determine Tube support plate intersections where tube deformations could potentially result in tube collapse during the combined LOCA and SSE event. The staff, therefore, finds TVA's analytical methodology acceptable. TVA gave the NRC a summary of the number of intersections at each Tube support plate that are excluded due to combined LOCA and SSE loadings. The licensee determined that it will not apply the voltage-based repair criteria to 466 intersections (out of about 100,588 total) on this basis. The top Tube support plate contains the largest number of excluded intersections (256) which represent 2.7 percent of the total hot and cold-leg intersections at that Tube support plate. Excluded tubes are clustered around wedge locations. A large number of the affected tubes contain multiple excluded intersections, both at multiple plate elevations and at both hot and cold-leg intersections. Hence, the total number of tubes affected is less than the number of affected intersection locations. The licensee has used bounding LOCA rarefaction wave loadings and Watts Bar Unit 1 seismic input data to develop the list of tubes provided in its submittal. TVA compared the applied loadings to data from a Tube support plate crush test program to ultimately define the list of excluded intersections. The NRC approved use of leak-before-break methodology for the Watts Bar Unit 1 loop piping, and use of small-break LOCA loadings is, therefore, justifiable, although not credited in this submittal. The staff finds this approach conservative and acceptable because TVA used large-break LOCA loadings in the analysis which bound the small-break locations. The intersections identified in the submittal are locations where the tube could potentially leak due to opening of the flaw during tube deformation. However, deformation potential does not imply that the tube will completely collapse, thereby resulting in a total loss of flow area during a LOCA and SSE event. In the staff's view, this provides additional conservatism in the tube exclusion analysis. The NRC has previously approved the use of small-break combined event loadings for ARC evaluation with similar steam generators at Byron Station, Units 1 and 2. Hence, if the licensee decides to perform a plant-specific tube exclusion analysis based on small break combined event loadings at some future date (as stated on page E1-5 of its submittal of April 10, 2000), the staff would find that approach acceptable in principle. However, TVA would still need to seek NRC review and approval of the amendment request.

3.3.1 Conditional Probability of Burst

In accordance with GL 95-05, TVA will perform probabilistic analyses, using the methodology described in Revision 1 of WCAP-14277, to quantify the potential for steam generator tube ruptures given a MSLB event. TVA will compare the results of the probabilistic analyses to a threshold value of $1x10^{-2}$ per cycle in accordance with GL 95-05. This threshold value assures that the probability of burst is acceptable considering calculation assumptions and the results of the staff's generic risk assessment for steam generators contained in NUREG-0844, "NRC Integrated Program for the Resolution of Unresolved Safety Issues A-3, A-4, and A-5 Regarding Steam Generator Tube Integrity." Failure to meet the threshold value indicates that ODSCC confined to within the thickness of the tube support plate could contribute a significant fraction to the overall conditional probability of tube rupture from all forms of degradation assumed and evaluated as acceptable in NUREG-0844. The NRC staff concludes TVA's proposed methodology for calculating the conditional burst probability is consistent with the guidance in GL 95-05 and is acceptable.

3.3.2 Effects of Pressurization Rate and Reinforcing Foil on Burst Pressure

An in situ pressure test of a degraded steam generator tube and subsequent laboratory testing of electric discharged machined (EDM) flaws of a similar geometry suggested that the rate of pressurization can influence measured burst pressure of a specimen. In response to this finding, the industry and TVA began an effort to develop an understanding of these test results and their potential implications on existing burst pressure databases and industry integrity assessment evaluation models.

The industry's effort was aimed at understanding the effect for all burst pressure databases (i.e., a generic assessment). TVA's effort was more aimed at understanding whether the effect impacted the databases they wanted to use in support of their license amendment request to implement the voltage-based tube repair criteria discussed in GL 95-05.

As a result, the following evaluation focuses on TVA's use of the 3/4-inch diameter GL 95-05 burst pressure database, and whether pressurization rate of the specimens or other similar phenomena significantly affects this database. Although the review focuses on TVA's license amendment application, the staff had to evaluate some of the generic assessment information to complete its review of TVA's submittal.

The industry conducted a burst testing program to determine the reason for the differences in the burst pressures observed from a flaw tested in situ (under a slow pressurization rate without foil reinforcement) and from subsequent laboratory testing of EDM flaws of a similar geometry (under a fast pressurization rate with foil reinforcement). A flaw, referred to as a Type 14 specimen, was selected for burst testing under the following three conditions:

- fast pressurization rate with foil
- slow pressurization rate without foil
- fast pressurization rate without foil

The Type 14 specimen was selected because of its similarity to the flaw for which the effect was initially observed. About 6 to 10 specimens were tested for each condition. The specimens were supposed to be identical; however, there were minor variations in the flaw geometries. To account for these variations, the industry normalized the measured burst pressures (dividing measured burst pressure by predicted (calculated) burst pressure based on the actual flaw dimensions). Table 1 gives the test results.

Table 1 - Normalized Measured / Calculated Burst Ratios for Various Test Conditions (Type 14 Specimens)

Test Condition	Mean (percent)	Standard Deviation (percent)
Fast Test With Foil	121.2	9.1
Slow Test Without Foil	97.6	4.2
Fast Test Without Foil	97.3	4.2

Since the slow and fast tests performed without foil reinforcement indicate the burst pressures of the specimens are nearly identical, the industry concluded the original effect observed was a foil-strengthening effect rather than a pressurization-rate effect.

The Type 14 specimen contains a relatively long (1.4-inch) flaw with two nearly through-wall portions. Under GL 95-05, the flaw length is limited to the thickness of the tube support plate (which is 0.75-inch). Additional tests were performed to determine if the use of foil reinforcement affected the burst pressure values in the GL 95-05 burst pressure database assuming the initially observed effect was entirely due to a foil effect rather than a pressurization rate effect. Approximately 20 percent of the GL 95-05 burst pressure database is believed to be obtained under slow pressurization rates. Reinforcing foil was used in both the slow and fast pressurization rate tests with the foil normally used for deeper cracks since deeper cracks may leak prior to bursting.

The second phase of testing was conducted with a variety of 0.75-inch flaws to determine if the foil effect was present in shorter flaws, which are the focus of GL 95-05. These flaws had a centrally deep section which measured 0.25-inch in length. The depth of this centrally deep section varied between approximately 55 percent and 100 percent through-wall. The depth on either side of this centrally deep section was gradually reduced until it reached approximately 50 percent through-wall at the ends of the 0.75-inch long flaw. This geometry was considered representative of flaws present at the tube support plate elevations to which the GL 95-05 repair criteria would be applied. Based on the results of these tests, TVA concluded there was no demonstrated foil strengthening effect on the 0.75-inch long flaw configurations although they did indicate there was a possibility that the scatter in the test data could mask a possible mild foil strengthening effect of several percent at most.

TVA did not conduct an exhaustive program to determine when (i.e., for what combination of flaw length and depth) the foil effect would be noticed. Rather, TVA provided a rationale for why an apparent large foil strengthening effect was observed in the burst tests of the 1.42-inch long Type 14 specimen, while the 0.75-inch long profiles used to examine the foil strengthening issue show, at most, a minor effect. TVA postulated the effect was due to the limited amount of opening that the remaining ligaments beneath very deep cracks (or EDM slots) can tolerate before tube burst (or early ligament failure) as discussed below.

For very deep cracks, the remaining ligament cannot deform to a large extent before tearing. Early ligament failure will occur if the strength of the ligament is not sufficient to limit the opening it must tolerate before a complete plastic collapse condition is reached. This can result in tube burst. The slot (or crack) width opening that a ligament can tolerate scales with its size. Smaller ligaments (i.e., more severe flaws) will tear at correspondingly smaller openings. Crack length and ligament size control the opening forced upon a ligament. In addition, the ligament material provides a closure force opposing the slot (or crack) opening. Thus, larger ligaments (i.e., less severe flaws) can both tolerate larger openings and act to limit the amount of opening. The ligament in a short crack is, in effect, helped by the unflawed adjacent material experiencing relatively little deformation. For a given ligament thickness, early ligament tearing is more prone to develop in a long crack than in a short crack. Thus, closure forces provided by the reinforcing foil can delay ligament failure, leading to an increase in overall strength. Since limited flaw opening is needed for failure for long, deep slots (or cracks), larger foil strengthening effects are expected for these flaws compared to shorter flaws with the same ligament size.

For the flaws of concern for GL 95-05 (i.e., flaws < 3/4-inch in length), TVA also determined there was no systematic effect of the pressurization rate on the burst pressure to voltage correlation for 3/4-inch diameter tubes. This evaluation was possible since approximately 20 percent of the database was from non-US sources whose testing practices frequently involved slow pressurization rates. This result is consistent with the results provided in the table above. In addition to these arguments, TVA indicated that no time-dependent effects were observed during leak-rate testing performed without a foil. These tests involved significant hold times and were performed at a differential pressure of about 3000 pounds per square inch and a temperature of approximately 600 °F.

Based on the information provided by TVA and from the NRC's Office of Nuclear Regulatory Research, the staff believes there is reasonable assurance that tube pressurization rate and/or foil effect do not significantly affect the 3/4-inch burst pressure database proposed by TVA. This is not to say that there is no time-dependent effect on burst pressure (e.g., pressurization rate effect) and/or there is no foil effect affecting the 3/4-inch diameter database. However, the staff finds that the effects, if any, are small. Furthermore, the staff concludes that future limitations on the pressurization rate of flaws (consistent with the industry's interim guidance on the conduct of pressure tests) and additional research will provide additional insights on these phenomena.

3.3.3 Accident Leakage

TVA will use the methodology described in Revision 1 of WCAP-14277 to calculate the steam generator tube leakage from the faulted steam generator during a postulated MSLB event. The methodology consists of the following two major components:

- a model predicting the probability that a given indication will leak as a function of voltage (i.e., the probability of leakage model)
- a model predicting leak rate as a function of voltage, given that leakage occurs (i.e., the conditional leak rate model)

The staff concludes that TVA's proposed methodology for calculating the tube leakage is consistent with the guidance in GL 95-05 and is acceptable.

3.3.4 Primary-to-Secondary Leakage

The voltage-based repair criteria would allow degraded tubes to remain in service. Therefore, the degraded tubes may develop through-wall cracks which may leak during normal operation, transients, or postulated accidents. As a defense-in-depth measure, GL 95-05 specifies that the operational leakage limits of the plant technical specifications be limited to 150 gallons per day (gpd) from any one steam generator. TVA proposed to change the leakage limits in the plant technical specifications to 150 gpd through any one steam generator. In addition, TVA has incorporated the guidelines in Electric Power Research Institute (EPRI) Report TR-104788, "PWR Primary-to-Secondary Leak Guidelines," into the Watts Bar plant operating procedures. The staff concludes that the proposed operational leakage limit of 150 gpd for Watts Bar technical specifications is consistent with GL 95-05 and, therefore, is acceptable.

3.4 Degradation Monitoring

To confirm the nature of the degradation at the tube support plate elevations, licensees periodically remove tubes from the steam generators for destructive testing. The test data from removed tubes is used to do the following:

- confirm that the nature of the degradation observed at these locations is predominantly axially oriented ODSCC
- provide data for assessing the reliability of the inspection methods
- supplement the existing databases (e.g., burst pressure, probability of leakage, and leak rate)

GL 95-05 specifies that at least two tubes be removed from steam generators with the objective of retrieving as many (minimum of four) intersections as practical during the plant steam generator inspection outage preceding initial application of the voltage-based repair criteria. On an ongoing basis, additional (minimum of two) tube specimens should be removed at the first refueling outage following 34 effective full-power months of operation or at the maximum interval of three refueling outages after the previous tube pull.

TVA proposed an alternative tube pull program to the above GL 95-05 guidelines. TVA's April 10, 2000, submittal stated that it will follow the industry-proposed guidance for tube removal in EPRI report, NP-7480-L, Addendum 3, "Steam Generator Tubing Outside Diameter Stress Corrosion Cracking at Tube Support Plates Database for Alternate Repair Limits, 1999 Database Update." The tube removal guidance in the EPRI report deviates from GL 95-05. The industry proposed delaying tube removal for one cycle if no indications $\geq$ 3 volts are found during tube inspection. In a letter to David J. Modeen of NEI dated January 31, 2000, the NRC staff found that some parts of the industry-proposed tube removal program were acceptable and some parts were not. David Modeen's letter of June 2, 2000, on "Steam Generator Degradation Specific Management Database, Addendum 3" responded to the NRC's letter. NEI agreed with the staff's finding and indicated it would pursue the unacceptable items. TVA stated that it will only implement those parts of the industry-proposed alternative tube pull program that the NRC staff has approved in NRC's letter of January 31, 2000. The NRC staff finds that TVA's proposed tube pull program is acceptable because it is consistent with the tube pull program approved by the NRC staff as discussed in the January 31, 2000, letter.

3.5 Changes to Technical Specifications

TS 3.4.13.d. — The total primary-to-secondary leakage through all steam generators is changed from 1 gallon per minute to 600 gpd.

TS 3.4.13.e. — The primary-to-secondary leakage through any one steam generator is changed from 500 to 150 gpd.

TS 5.7.2.12.b.2.e) — A section is added to specify that bobbin probes will be used to inspect all indications left in service at the flow distribution baffles and tube support plate elevations during all future refueling outages.

TS 5.7.2.12.e. — A paragraph is added to require a 100 percent bobbin coil inspection for hot-leg and cold-leg tube support plate intersections (including the flow distribution baffles) down to the lowest cold-leg tube support plate with known 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 tube inspected over their full length.

TS 5.7.2.12.g.1.f) — A paragraph is added to specify that the 40 percent through-wall plugging limit does not apply to indications detected at the flow distribution baffle and tube support plate intersections for which the voltage-based repair criteria are being applied.

TS 5.7.2.12.g.1.l) — A section is added to define the tube support plate repair limit. This section includes various definitions of voltage-based criteria in accordance with GL 95-05.

TS 5.9.9 — This section is modified to be consistent with the reporting requirements specified in GL 95-05.

Bases 3.4.13 — In the Background section of the Bases, a paragraph is added to describe the upper voltage limit in accordance with GL 95-05.

Bases 3.4.13.d and 3.4.13.e — The Bases is modified to include the information regarding primary-to-secondary leakage limits to be consistent with the guidelines in GL 95-05.

The staff finds that the proposed changes to the Watts Bar TS follow the guidance in GL 95-05 and are, therefore, acceptable.

3.6 Summary

The staff has reviewed TVA's proposed amendment to implement the voltage-based repair criteria for steam generator tubes in the technical specifications for Watts Bar Unit 1. The staff concludes that the proposed alternate repair criteria are consistent with GL 95-05 and are acceptable. The staff also concludes that adequate structural and leakage integrity of steam generator tubing can be assured, consistent with 10 CFR Part 50 requirements, for indications to which the voltage-based repair criteria will be applied. The staff approves the proposed voltage-based repair criteria based, in part, on TVA being able to successfully demonstrate after each inspection outage (as shown in its 90-day steam generator tube inspection report) 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. On the basis of the staff conclusions, TVA may incorporate the proposed alternate repair criteria into the technical specifications for Watts Bar Nuclear Plant, Unit 1.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Tennessee State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (66 FR 34751). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

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

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