August 15, 2003

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 AN AMENDMENT FOR STEAM GENERATOR TUBE REPAIR (TAC NO. MB6976)

Dear Mr. Scalice:

The Commission has issued the enclosed Amendment No. 44 to Facility Operating License No. NPF-90 for Watts Bar Nuclear Plant (WBN), Unit 1. The amendment consists of changes to Technical Specification (TS) 5.7.2.12, "Steam Generator (SG) Tube Surveillance Program," and is in response to your application dated December 13, 2002, as supplemented on May 19 and July 11, 2003.

The revised TS would allow the use of Westinghouse leak-limiting Alloy 800 sleeves to repair defective SG tubes as an alternative to plugging the tube.

A copy of the safety evaluation is also enclosed. Notice of issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Margaret H. Chernoff, 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. 44 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. 44
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 December 13, 2002, as supplemented on May 9, 2003, and on July 9, 2003, 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. 44, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. TVA shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Allen G. Howe, 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: August 15, 2003

ATTACHMENT TO AMENDMENT NO. 44

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 area of change.

Remove Pages	Insert Pages
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	5.0-19a
5.0-19b	5.0-19b
5.0-19c	5.0-19c
5.0-20	5.0-20
5.0-20a	5.0-20a
5.0-21	5.0-21

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 44 TO FACILITY OPERATING LICENSE NO. NPF-90

TENNESSEE VALLEY AUTHORITY

WATTS BAR NUCLEAR PLANT, UNIT 1

DOCKET NO. 50-390

1.0 INTRODUCTION

By letter dated December 13, 2002, as supplemented by letters dated May 19 and July 11, 2003, (ADAMS Accession Nos. ML023520628, ML031430185, and ML032020110, respectively), the Tennessee Valley Authority (TVA or the licensee), submitted a request for changes to the Technical Specifications (TSs) for Watts Bar Nuclear Plant (WBN), Unit 1. The requested changes would modify TS 5.7.2.12, "Steam Generator (SG) Tube Surveillance Program," to allow the use of Westinghouse leak-limiting Alloy 800 sleeves to repair defective steam generator tubes as an alternative to plugging the tube.

Currently, the TSs allow defective tubes to be removed from service by installing plugs at both ends of the tube. The proposed amendment would provide an alternative to plugging defective SG tubes. By repairing the defective tubes with the use of leak limiting sleeves, the tube is allowed to remain in service. A sleeve is a tube segment with a diameter smaller than the parent tube. The sleeve is inserted into and expanded inside the parent tube to form a structural joint. The licensee proposes a tube repair method that uses Westinghouse leak-limiting sleeves made of Alloy 800 material. The Alloy 800 sleeve is not required to be leak tight. The design, installation, analysis, and qualification tests of the sleeve are documented in the Westinghouse topical report, "Steam Generator Tube Repair for Combustion Engineering and Westinghouse Designed Plants with 3/4 Inch Inconel 600 Tubes Using Leak Limiting Alloy 800 Sleeves," WCAP-15918-P, Revision 00, dated November 2002 (proprietary).

The supplemental letters provided clarifying information that did not expand the scope of the original request or change the initial proposed no significant hazards consideration determination.

2.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix A, General Design Criterion (GDC) 14, Reactor coolant pressure boundary, contains requirements applicable to steam generator tubes. GDC 14 requires that the reactor coolant pressure boundary be

designed, fabricated, erected, and tested in order to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. Steam generator tubes represent the majority of the reactor coolant pressure boundary. When a tube is defective, the tube is either repaired using a sleeve or removed from service by plugging. To repair a part of the existing reactor coolant pressure boundary, the sleeve should be qualified for service in accordance with the specifications in Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, which refers back to Section III of the ASME Code. The original SG tubes are designed in accordance with Section III of the ASME Code. The sleeve repairs the degraded tube(s); therefore, Section III is applicable to sleeve design. The sleeve should be analyzed by appropriate ASME Code equations considering design, operating, and accident loading conditions. The resulting sleeve stresses should satisfy corresponding ASME Code limits. In addition, the sleeve wall thickness must satisfy the minimum wall thickness requirement of the ASME Code.

10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," requires a quality assurance program for the design, fabrication, construction, and operation of structures, systems, and components in nuclear plants. The pertinent requirements of Appendix B apply to all activities affecting the safety-related functions of those structures, systems, and components. These activities include designing, purchasing, fabricating, handling, shipping, storing, cleaning, erecting, installing, inspecting, testing, operating, maintaining, repairing, refueling and modifying safety-related structures, systems and components.

Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded Pressurized Water Reactor Steam Generator Tubes," provides guidance for determining minimum wall thickness, beyond which the degraded tube should be plugged (i.e., plugging limits). RG 1.121 also provides performance criteria that recommend that the margin of safety against tube rupture under normal operating conditions should not be less than 3 at any tube location where defects have been detected. The margin of safety against tube failure under postulated accidents, such as a loss-of-coolant accident, steam line break, or feedwater line break concurrent with the safe shutdown earthquake, should be consistent with the margin of safety determined by the stress limits specified in Section III of the ASME Code.

3.0 TECHNICAL EVALUATION

The staff reviewed the following areas of the proposed tube repair method: design, installation, material selection, qualification tests, structural analyses, and nondestructive examination of the Westinghouse Alloy 800 sleeve. The staff also reviewed the proposed changes to TS 5.7.2.12 to determine the acceptability of the tube repair implementation. These topics are discussed below:

3.1 Design

Two leak-limiting Alloy 800 sleeve designs were proposed for use in repairing a tube. A transition zone sleeve is designed to repair tube degradation in the vicinity of the top of the tubesheet. A tube support sleeve is designed to repair tube degradation at the tube support plate intersections or in the freespan region. The length of the transition zone and tube support sleeves is sized to encompass the degraded regions of the tube. To attach the sleeve to the parent tube, the transition zone sleeve uses several hydraulic expansion joints in the upper

region of the sleeve and a hard roll joint in the lower region of the sleeve that is located inside the tubesheet. The tube support sleeve uses several hydraulic expansion joints in the upper and lower regions of the sleeve.

3.2 Installation

Prior to sleeve installation, the inside surface of the candidate tube is cleaned and a visual inspection (VT) is performed on every candidate tube to verify the quality of the contact surface. Westinghouse indicated that once the cleaning process has been demonstrated to ensure appropriate cleaning, a sampling VT of cleaned tubes may be used. After VT, the sleeve is inserted into the cleaned tube and positioned at the desired location. An expansion device is inserted into the sleeve to expand the sleeve to make the required number of hydraulic expansion joints with the parent tube. The expansion device is controlled and monitored to ensure consistent diametral expansion. The hydraulic expansion joints support the required structural and leakage integrity while minimizing residual stresses in the parent tube. The hard roll in the lower end of the transition zone sleeve is performed by a roll expander. The torque of the roll expander is also monitored and controlled. After the installation, all sleeve-tube joints will undergo an initial acceptance and baseline inspection using an eddy current technique. The licensee will plug any sleeve if (1) an unacceptable set of expansions occurred, (2) the torque value for the rolled joint did not fall within the proper torque range, (3) the two sets of expanded joints were not positioned at the proper elevation, or (4) an unacceptable indication is found in the pressure boundary of the tube/sleeve assembly.

3.3 Materials of Selection

The sleeve material, Alloy 800, is a nickel-iron-chromium alloy. Westinghouse selected Alloy 800 for its favorable mechanical properties and corrosion resistance in both the primary and secondary side water chemistry. It is procured to the requirements of the ASME Code, Section II, Part B, SB-163, NiFeCr Alloy, Unified Numbering System (UNS) N08800, and Section III, Subsection NB-2000. Alloy 800 is incorporated in ASME Code Case N-20 and is considered acceptable for use by Regulatory Guide 1.85, “Materials Code Case Acceptability ASME Section III, Division 1,” Revision 24, dated July 1986. Westinghouse also requires additional restriction on contents of various chemical elements and specifies a certain annealing temperature and yield strength for the Alloy 800 sleeve.

3.4 Qualification Testing

Westinghouse performed the qualification tests in accordance with Appendix B to 10 CFR Part 50. The testing program included mechanical load tests, leakage tests, and corrosion tests. The mechanical load tests included axial load tests, pressure tests, collapse tests, and load cycling tests. The tests were performed on sleeve-tube mock-ups that were constructed to the same dimensions as the installed sleeves in the field.

Westinghouse performed the axial load tests to determine the structural integrity of the sleeve-tube joint under differential thermal expansion of an Alloy 800 sleeve and Alloy 600 tube. The test loads include the full range loading under startup, transient, normal power, shutdown, and accident conditions. The axial load tests showed that the Alloy 800 sleeve can support differential thermal conditions and accident loads even if the parent tube is severed.

Westinghouse performed the pressure tests to determine the structural integrity of the sleeve-tube joint under primary-to-secondary pressure differentials during normal operating, transient and postulated accident conditions. The pressure tests showed that the sleeve-tube joint will maintain a margin of 3 with respect to the normal operating differential pressure load as specified in RG 1.121.

Westinghouse performed the collapse tests to show that the sleeve would not collapse if water is trapped in the annulus region between the inside surface of the parent tube and the outside surface of the sleeve. The trapped water may be pressurized during operation and potentially cause the sleeve to collapse. The collapse tests showed that the sleeve would not collapse under the maximum secondary side pressure.

Westinghouse performed the load cycling tests to show that the structural and leakage integrity of the sleeve-tube joint will be maintained under cyclical differential thermal expansion and internal pressure in normal operating and transient conditions. The load cycling tests included fatigue tests, thermal cycling tests, and mechanical load cycling tests. The load applied in the cycling tests was greater than three times the maximum differential pressure. These tests showed that under various temperatures, the sleeve-tube joint is not degraded by cyclic loads. The cycling tests confirm that slip during the initial heat-up is small, and the sleeve repositions itself inside the parent tube to accommodate the thermal expansion without subsequent slip. As a part of the load cycling tests, the specimens were also tested for leakage integrity. The leak tests showed that the seal in the hydraulically expanded joints improved after load cycling. Westinghouse performed leak rate tests on the sleeve-tube assembly for various temperatures and pressures under normal operating and main steam line break conditions. The test results showed that the leakage from a single sleeve is extremely small relative to the operational primary-to-secondary leakage limit in the plant technical specifications and the allowable leakage under accident conditions (see discussion below). It would take thousands of leaking sleeves to reach either operational or accident condition leakage limits.

Sleeve-tube joints increase the residual stresses in the parent tube which, in turn, may cause the tube to be susceptible to stress corrosion cracking. Westinghouse stated that the Alloy 800 sleeve is designed to impart minimal residual stresses to the parent tube to avoid potential corrosion in the hydraulic expansion joints. Westinghouse has performed various corrosion tests and assessments of Alloy 800 sleeves with full-length sleeved-tube mock-ups. Sleeve-tube assemblies were pressurized with highly corrosive solutions. Westinghouse also performed the corrosion tests to assess the relative time to cracking of the sleeve-tube joint. Alloy 800 sleeve did not develop any cracking in either the primary or secondary side tests. The Alloy 800 sleeve has demonstrated higher corrosion resistence than the Alloy 600 parent tube.

Westinghouse stated that the Alloy 800 sleeves have not had service induced degradation or significant leakage in nuclear plants. Westinghouse also stated that besides Alloy 800 sleeves, Alloy 800 tubing has been used in pressurized-water reactor conditions in international nuclear plants without experiencing significant primary or secondary side stress corrosion cracking. This is based on experience of more than 200,000 tubes in service for up to 19 years. However, the staff judges that the time for the initiation of corrosion in sleeve-tube assemblies is difficult to accurately quantify. Although vendors traditionally conduct accelerated corrosion tests of sleeve-tube assemblies to predict service life, the staff finds this method is unreliable for deterministic predictions. The staff does consider that the corrosion tests give a viable indicator of potential performance. Presently, the staff can only assume a limited life expectancy for

Alloy 800 sleeves. Considering the uncertainties in sleeve life expectancy, the staff requires licensees to inspect a sample of sleeves at each refueling outage, as discussed below, to ensure that any degradation in the sleeve assembly is detected and addressed early.

3.5 Inspection

Westinghouse specified that the +Point coil probe will be used in the inservice inspection of Alloy 800 sleeves. The eddy current technique has been qualified in accordance with Appendix H of the Electric Power Research Institute (EPRI) report, "Pressurized Water Reactor Steam Generator Examination Guidelines," Revision 5, September 1997. Westinghouse qualified the sleeve inspection technique on actual sleeve-tube assemblies that included electro-discharge-machining notches and drilled flaws at each of the transition and expansion zones. The licensee proposed an initial, minimum inspection of 20-percent sleeves at each refueling outage as shown in the proposed change to the TSs. In addition to the 20-percent initial sample, additional sleeves may be inspected depending on the extent of sleeve degradation. The proposed inspection sampling is consistent with the current industry guidance for steam generator sleeve examinations as specified in the EPRI report. In the December 13, 2002, letter, the licensee stated that future inservice inspection will be consistent with plant TSs and Revision 6 of the EPRI guidance. In its response to the staff's request for additional information, the licensee stated that the primary difference between Revision 5 and Revision 6 of the EPRI guidance is the separation of the various types of material and how each is inspected.

For degraded sleeves, TVA will implement a plug-on-detection approach as shown in the proposed TSs. This is consistent with Westinghouse's recommendation in the topical report. The staff finds this disposition approach conservative and, therefore, acceptable.

3.6 Structural Analyses

Westinghouse performed structural analyses in accordance with 10 CFR Part 50, Appendix B, and Section III of the ASME Code. The structural analyses included applied loads from pressure, relative displacement, fatigue, axial, seismic, and thermal radial differential under normal and accident loading conditions. In the analyses, Westinghouse assumed two bounding tube configurations: (1) the tube is intact, and (2) the tube is severed at the flaw location. In addition, Westinghouse assumed two bounding tube-support plate configurations: (1) the tube is free to move past the tube support plates, and (2) the tube is locked in the first tube support plate and is prevented from axial motion. The structural analyses showed that stresses and fatigue factor in the worst sleeve-tube configuration satisfy the allowables in Section III of the ASME Code.

Westinghouse's structural analysis also included calculations for minimum required sleeve thickness based on ASME Code Section III, Paragraph NB-3324.1. The calculations show that the actual sleeve wall thickness is greater than the minimum required thickness, and, therefore, is structurally acceptable. Westinghouse also calculated the percentage of sleeve wall thickness for allowable degradation, considering cracking and wall thinning, as specified in RG 1.121. The cracking model gives a more conservative degradation allowable than the wall thinning model. As discussed above, TVA proposed a plug-on-detection approach which will not allow any degraded sleeves to remain in service. This strategy is conservative and is, therefore, acceptable.

Under severe accident conditions in which primary system temperature may reach 1200 to 1500 degrees F, the material properties of Alloy 800 are not significantly different from that of Alloy 600 at the severe-accident temperatures. The staff concurs that the severe-accident temperature does not significantly affect the structural integrity of the pressure boundary of Alloy 800 sleeve as compared to Alloy 600 parent tubing.

3.7 Leakage Integrity

Westinghouse has determined the sleeved joint leakage via laboratory testing to be small. For the leakage integrity assessment methodology, the licensee will conservatively assume all installed sleeves leak and will use the leak rate at the upper 95 percent confidence limit on the mean value of leakage for appropriate temperature and pressure conditions. The licensee will combine the total sleeve leak rate with the total amount of leakage from all sources for comparison against the limit on accident induced leakage as specified in the Updated Final Safety Analysis Report (UFSAR) Section 15.5.4, "Environmental Consequences of a Postulated Steam Line Break," and UFSAR Table 15.5-7, "Dose from Main Steam Line Break." In addition, primary-to-secondary operational leakage is limited to 150 gallons per day through any one steam generator in the current Watts Bar TS 3.4.13.e. The staff finds that the leakage tests have demonstrated the adequacy of limiting leakage capacity of the sleeve. The staff also finds the licensee's leakage integrity assessment methodology conservative and is, therefore, acceptable.

3.8 Summary

The licensee has performed structural analyses and tests for a variety of loadings that enveloped plant-specific design, operating, transient and accident loads. The analyses, testing, and operating experience demonstrate that the leak-limiting Alloy 800 sleeve is capable of restoring steam generator tube integrity. The staff finds that the licensee has demonstrated the acceptability of the sleeve repair in accordance with Appendix B to 10 CFR Part 50, GDC 14 of Appendix A to 10 CFR Part 50, RG 1.121, and the ASME Code. The staff concludes that the licensee may incorporate the proposed changes into the Watts Bar Unit 1 TSs.

4.0 Proposed Changes to Technical Specifications

The licensee proposed the following changes to TS 5.7.2.12, "Steam Generator (SG) Tube Surveillance Program":

1. TS 5.7.2.12.g.1.e is revised as follows:

> Imperfection - An exception to the dimensions, finish, or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, are to be considered as imperfections;

2. TS 5.7.2.12.g.1.f is revised by adding the following requirement:

> Plugging Limit - The imperfection depth at or beyond which the tube shall be removed from service by plugging, or repaired by sleeving, in the affected area. The plugging and repair criteria are specified as follows:
>
> A sleeved tube shall be plugged if an imperfection is detected in a Westinghouse Alloy 800 leak-limiting sleeve.
>
> A sleeved tube shall be plugged if an imperfection is detected in the pressure boundary portion of the original tube wall in the Westinghouse Alloy 800 leak-limiting sleeve/tube assembly (i.e., at the sleeve-tube joint(s)).
>
> A tube shall be plugged or repaired if the depth of an imperfection in the original tube wall is greater than or equal to 40% of the nominal wall thickness. This definition does not apply to imperfections detected in the non-pressure boundary portion of the original tube wall associated with a sleeve. This definition does not apply to the portion of the original tube in the tubesheet below the F* distance provided the tube does not have a sleeve installed in the tubesheet region and the tube is not degraded within the F* distance.

3. TS 5.7.2.12.g.1.m is added to the TSs as a new requirement.

> Tube Repair refers to a process that reestablishes tube serviceability. Tube repair of defective tubes will be performed by installation of the Westinghouse Alloy 800 leak-limiting repair sleeve as described in the proprietary Westinghouse Report WCAP-15918-P, Revision 00 (Draft CEN-633-P, Revision 05-P), “Steam Generator Tube Repair for Combustion Engineering and Westinghouse Designed Plants with 3/4 Inch Inconel 600 Tubes Using Leak Limiting Alloy 800 Sleeves.”

4. Table 5.7.2.12-1 is modified to include the option of repairing defective tubes.

5. Table 5.7.2.12-2 is added to the TSs as a new requirement.

6. Reference to “Table 5.7.2.12-2" is added to the following TS sections: TS 5.7.2.12.a; TS 5.7.2.12.b; TS 5.7.2.12.d; TS 5.7.2.12.f.2 and f.3; and TS 5.7.2.12.g.2.

7. Old Table 5.7.2.12-2, and references to it, are renumbered to be Table 5.7.2.12-3 in the revised TS.

The staff finds that the proposed changes are consistent with the technical basis of the proposed sleeve repair method and are, therefore, acceptable.

5.0 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.

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes requirements 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 (68 FR 12958). 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.

7.0 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.

Principal Contributor: J. Tsao

Date: August 15, 2003

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Tennessee Valley Authority

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