TVA

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

Watts Bar Nuclear Plant, Unit 2
NRC Docket No. 50-391

Subject: Watts Bar Nuclear Plant (WBN) Unit 2 – Status of Unresolved Safety Issues

References:
1. Safety Evaluation Report Related to Operation of Watts Bar Nuclear Plant Units 1 and 2, Docket Numbers 50-390 and 50-391, June 1982 including Supplement 3, January 1985, Supplement 4, March 1985, Supplement 7, September 1991, Supplement 8, January 1992, Supplement 9, June 1992, Supplement 11, April 1993, Supplement 13, April 1994, Supplement 14, December 1994, and Supplement 15, June 1995

2. TVA to NRC letter dated December 2, 2010, "Watts Bar Nuclear Plant (WBN) Unit 2 – Unresolved Safety Issue Status (A-12 and A-36)"

In response to a verbal request from NRC Project Management, this letter provides, in the enclosure, the WBN Unit 2 implementation plan for Unresolved Safety Issues. Previously, in Reference 2, the status of A-12 and A-36 was submitted. That status is repeated here with some additional information for completeness. There are no new commitments associated with this submittal. If you have any questions, please contact William Crouch at (423) 365-2004.

Respectfully,

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Enclosure: WBN Unit 2 implementation plan for Unresolved Safety Issues cc (Enclosure):

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Enclosure

Watts Bar Nuclear Plant (WBN)
WBN Unit 2 Implementation Plan for Unresolved Safety Issues

In response to a verbal request from NRC Project Management, this letter provides the WBN Unit 2 implementation plan for Unresolved Safety Issues (USIs). Previously, in Reference 2, the status of A-12 and A-36 was submitted. That status is repeated here with some additional information for completeness. The status is given in two sections below. Section A contains issues closed in the original Safety Evaluation Report (SER). Section B contains the remainder of the issues and the applicable supplements to the SER.

A. The following items were closed for WBN in the original SER (NUREG-0847, Appendix C). A review has been performed, and the results ensure the basis for the original closure remains valid for Unit 2.

A-2 Asymmetric Blowdown Loads of PWR Primary System

The Unit 1 and Unit 2 reactor vessels and reactor coolant systems are of an identical design, and the Unit 2 components will be operated at the same thermal-hydraulic conditions; therefore, the basis for closure remains valid for Unit 2.

A-9 Anticipated Transients Without Scram (ATWS) of Light Water Reactors

The SER evaluated the ATWS Mitigating System Actuation Circuitry (AMSAC) design for both Unit 1 and Unit 2 and found them to be acceptable. In September 2000, Unit 1 replaced the digital system with a relay based analog system. This change was performed under 10 CFR 50.59. Unit 2 is installing a similar relay based analog system. The Unit 1 hardware change resulted in no changes in the setpoints or operation of the Unit 1 AMSAC circuitry. Therefore, the basis for closure remains valid for Unit 2.

A-24 Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment

Unit 2 provides the environmental qualification (EQ) of electrical equipment to the same requirements as used for Unit 1. The Unit 2 procedures likewise implement the WBN EQ process; therefore, the basis for closure remains valid for Unit 2.

A-26 Reactor Vessel Pressure Transient Protection for Pressurized Water Reactors

Unit 2 contains the same design safety valves and power operated relief valves (PORVs) used on Unit 1. For Unit 2, the Unit 1 Cold Over Pressure Mitigation System (COMS) analog actuation circuitry has been replaced by a digital distributed control system (DCS). The DCS uses the same inputs and duplicates the function of the Unit 1 analog controls in software. As described in Unit 2 FSAR Amendment 102, Sections 7.7.1.11 and 5.2.2.4.1, the DCS provides redundant processors and other enhancements which improve COMS reliability. The valves and actuation circuitry will be tested in the same manner as used in Unit 1; therefore, the basis for closure remains valid for Unit 2.

A-31 Residual Heat Removal System

The Unit 2 Residual Heat Removal (RHR) system is designed the same as Unit 1, and thus has the same capability to cool the reactor to cold shutdown; therefore, the basis for closure remains valid for Unit 2.

B. The following items were closed for WBN in NUREG-0847 via SSERs as noted for each item. A review has been performed, and the results ensure the basis for the original closure remains valid for Unit 2.

A-1 Water Hammer

This item was closed generically by the NRC in NUREG-0927. SSER 15 provided an evaluation of Units 1 and 2 and reviewed the practices related to piping design and found them acceptable. Unit 2 continues to utilize the same design principles and transient evaluations as used for Unit 1. The Unit 2 pre-operational test program will demonstrate acceptable piping vibration in accordance with the requirements of Regulatory Guide 1.68. The SER evaluated the concern of operation of the Upper Head Injection (UHI) system and concluded that it was acceptable. Later, the Unit 1 UHI system was removed as has been done for Unit 2; therefore, the basis for closure remains valid for Unit 2.

A-3 Westinghouse Steam Generator Tube Integrity

Unit 2 utilizes model D-3 steam generators as reviewed in SSER 4 with no changes to the steam generator tubes; therefore, the basis for closure remains valid for Unit 2. Further, Unit 2 adopted Steam Generator Tube Integrity TSTF-442 in the Unit 2 Technical Specifications.

A-11 Reactor Vessel Materials Toughness

This is a Unit 1 specific issue as stated on Page 5-10, Section 5.3.1.1, of the original NUREG-0847 SER of June 1982. This issue was subsumed by equivalent margin analysis, which was evaluated and closed for Unit 1 in Section 5.3.1.1.1 of SSER 14. This USI is thus closed.

A-12 Potential for Low Fracture Toughness and Lamellar Tearing on PWR Steam Generator and Reactor Coolant Pump Supports

This item is based on the requirements of NUREG-0577 of the same title and was addressed in WBN Units 1 and 2 Calculation WBNSSG6-002. This calculation documents the results of the review of the steam generator and reactor coolant pump support materials utilizing Appendix II of Electric Power Research Institute (EPRI) Report NP-3528, “Requirements and Guidelines for Evaluating Component Support Materials Under Unresolved Safety Issue A-12.” Based on the results of this review, the steam generator and reactor coolant pump support materials (except for three heats of ASTM A564 Type XM16 bolts) comply with NUREG-0577 requirements, and the potential for low toughness fracture of these materials has been adequately assessed. The three heats of ASTM A564 Type XM16 bolts (i.e., heats 91251, 91081, and 91243) were used in upper steam generator support bolting applications. These heats were evaluated for adequate fracture toughness in Nonconformance Report (NCR) GENNEB8201. Corrective action for this NCR initiated a bolting reheat treatment program for both Units 1 and 2, which has been completed. NRC Inspection Report 50-390/84-03 and 50-391/84-03 dated February 15, 1984, closed this bolting item for Units 1 and 2.

Subsequently, Unit 1 experienced steam generator 1 and 4 upper lateral splice plate support bolting fractures during the first and second cycle of operation due to the improper installation of the bolts. The bolts were installed at a slight angle giving single point contact that caused the failure. The problem was resolved using an angled washer. This was a Unit 1 only issue.

A-17 Systems Interaction In Nuclear Plants

This item was closed generically based on Generic Letter 89-18 in SSER 15 and thus remains closed for Unit 2.

A-36 Control of Heavy Loads at Nuclear Power Plants

This item was closed in Supplement 13, Supplemental Safety Evaluation Report (SSER) Section 9.1.4, for Unit 1. Unit 1 performed the following actions:

1. Safe load paths (SLP) were clearly defined.
2. Load handling procedures were put into place.
3. Inspection and testing programs for overhead handling systems, including lift devices, were established.
4. Operator qualification and training are controlled and documented.
5. American National Standards Institute (ANSI) N14.6 governs the design and application of special lifting devices.
6. ANSI B30.9 governs those lifting devices assembled from manufacturer's components.
7. Crane design meets the requirements of Crane Manufacturers Association of America (CMAA) standard CMAA-70, ANSI B30.2, or other applicable industry standards. (Note: Supplement 13 inadvertently used CMMA-80 instead of CMMA-70.)

Directly related to this USI is Generic Letter 81-07 that requested a review of controls for the handling of heavy loads to determine the extent to which the guidelines of NUREG-0612 are being satisfied and to identify any changes and modifications that would be required in order to fully satisfy these guidelines. For USI A-36 and Generic Letter 81-07, Unit 2 will utilize the same approach used for Unit 1 including implementation of the guidelines of NEI 08-05 Rev. 0, Industry Initiative on the Control of Heavy Loads, as endorsed by the NRC per RIS 2008-08.

A-40 Seismic Design Criteria

This issue addresses the design of the refueling water storage tank. The Unit 1 and Unit 2 tanks were designed using the same criteria and design. The NRC closed the issue for WBN Unit 1 and 2 in SSER 7. The Unit 2 tank has not been modified; therefore, the basis for closure remains valid for Unit 2.

A-43 Containment Emergency Sump Reliability

SSER 9 discussed the sump design. Unit 2 is utilizing the Unit 1 program and design changes as the basis for closure of this issue. The containment emergency sump screens are being replaced with new screens which are as large as the Unit 1 screens but employ an enhanced internal flow pattern and thus results in improved performance. Unit 2 will have a conservatively lower fibrous material loading than Unit 1; therefore, the basis for closure remains valid for Unit 2.

A-44 Station Blackout

As addressed in SSER 13, the station blackout load profile has been changed to 4 hours with manual load shedding after 30 minutes into the event. Battery sizing for this load profile has been completed. The NRC documented its station blackout review in a letter dated March 18, 1993. The staff submitted supplement to that SER on September 9, 1993. The WBN site (Units 1 and 2) will maintain the 4-hour coping duration. Battery sizing will be demonstrated to be acceptable for two units. Unit 2 will implement battery load shedding procedures similar to Unit 1 procedures; therefore, the basis for closure remains valid for Unit 2.

A-45 Shutdown Decay Heat Removal Requirements

As addressed in SSER 15, this action was tracked for Unit 1 under the Individual Plant Evaluation (IPE) program. Unit 2 has an IPE program for dual unit operation that will utilize the same criteria used previously for Unit 1 except it will consider dual unit operation. Completion of the dual unit IPE will address the requirements of A-45; therefore, the basis for closure remains valid for Unit 2.

A-46 Seismic Qualification of Equipment in Operating Plants

As addressed in SSER 3, the scope of A-46 is limited to dealing with the seismic qualification of equipment in currently operating plants. The staff's evaluation of the WBN seismic qualification of equipment is discussed in Section 3.10 of the SER. The evaluation will not be handled under A-46 because it is being handled on a case by case basis; therefore, the open item identified under this USI should be deleted. The Unit 2 seismic qualification program is the same as Unit 1; therefore, the basis for deletion remains valid for Unit 2.

A-47 Safety Implications of Control Systems

As addressed in SSER 15, this issue was resolved for Unit 1 based on TVA’s letter dated October 24, 1990, and the draft Technical Requirements Manual (TRM). The Unit 2 TRM contains the same information as included in the current Unit 1 TRM; therefore, the basis for closure remains valid for Unit 2.

A-48 Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment

As addressed in SSER 8, SSER 4, and the SER, Unit 2 utilizes the same design hydrogen igniters (including backup power supply), containment spray system, and containment structure as Unit 1; therefore, the basis for closure remains valid for Unit 2.

A-49 Pressurized Thermal Shock

Generic resolution of this issue was effected by issuance of the final rule, 10 CFR 50.61; Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials"; and Generic Letters 88-11 and 92-01. The issue was resolved for WBN Units 1 and 2 by letter, S. C. Black to O. D. Kingsley, June 29, 1989 (regarding Generic Letter 88-11); Section 5.3.1, "Reactor Vessel Materials", of SSER 11; Section 5.3.1 of SSER 14 (regarding 10 CFR 50 Appendix G and Generic Letter 92-01). Unit 2 uses the same Regulatory Guidance; therefore, the basis for closure remains valid for Unit 2.