Tennessee Valley Authority, Post Office Box 2000, Spring City, Tennessee 37381-2000

June 27, 2011

10 CFR 50.4
10 CFR 2.390(b)(4)

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 – Response to Request for Additional Information Regarding April 27, 2011 NRC Audit

References:

1. NRC letter to TVA dated April 27, 2011, "Watts Bar Nuclear Plant, Unit 2 - Audit Report of Westinghouse Documents Relating to Final Safety Analysis Report Accident Analyses (TAC NO. ME4620)"
2. TVA letter to NRC dated November 9, 2010, "Watts Bar Nuclear Plant (WBN) Unit 2 - Final Safety Analysis Report (FSAR) - Response to Request for Additional Information"

During preparation for an audit the week of June 27, 2011, TVA determined the enclosed information should have been provided in response to Reference 1. Accordingly, the information requested in Reference 1, Section A.1 actions 4 and 5 is provided herein. Also, it should be noted that Reference 2, Attachment 6 provided the information requested by Reference 1, Section A.1, action 1.

If you have any questions, please contact Bill Crouch at (423) 365-2004.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 27th day of June 2011.

Respectfully,

David Stinson
Watts Bar Unit 2 Vice President

D030
NRR

Enclosure:

1. Supplemental Information for the Watts Bar Unit 2 (WBT) Hot Zero Power (HZP) Steam Line Break (SLB) Analysis for RAI Response

cc (Enclosure):

U.S. Nuclear Regulatory Commission
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Enclosure
Watts Bar Nuclear Plant (WBN) Unit 2 – Response to Request for Additional Information (RAI) Regarding April 27, 2011 NRC Audit

Supplemental Information for the Watts Bar Unit 2 (WBT) Hot Zero Power (HZP) Steam Line Break (SLB) Analysis for RAI Response

Reference: Westinghouse Letter WBT-D-3109

NRC Question A.4:

Address the staff's question, raised during the audit, regarding fuel rod mechanical design rod internal pressure (i.e., clad liftoff) concerns during the steamline break event down to the accumulator injection set pressure.

TVA Response:

The fuel design limit for rod internal pressure is that:

1. The internal pressure of the lead rod in the reactor will be limited to a value below that which could cause (1) the diametral gap to increase due to outward cladding creep during steady-state operations and (2) extensive DNB propagation to occur,
2. DNB propagation is limited for Condition II events, and
3. The increased fuel pressure does not introduce a significant number of additional DNB events in the safety analysis of Condition III and IV events.

Cycle-specific evaluations are performed using NRC-approved models and methodologies in order to confirm that all of the above criteria are met. These evaluations include analyzing the effects of steamline break events down to the accumulator injection set pressure.

NRC Question A.5:

What, if any, actions would plant operators initiate if the moderator temperature coefficient (MTC) surveillance measurement was within the most-negative allowable MTC, but beyond the cycle specific moderator density coefficients used in the steamline break analysis?

TVA Response:

There are no separate, tiered actions for Technical Specification requirements and cycle-specific values. The required actions are defined by Technical Specifications Limiting Condition for Operation (LCO) 3.1.4. Related information is provided in Reference 5 (WCAP-15088, Rev. 1) of the Core Operating Limits Report (COLR) section of the Watts Bar Unit 2 Technical Specifications (Section 5.9.5).