TVA

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

JUN 30 1994

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Gentlemen:

In the Matter of the Application of	)	Docket Nos. 50-390
Tennessee Valley Authority	)	50-391

WATTS BAR NUCLEAR PLANT (WBN) - CLARIFICATION OF INFORMATION IN FINAL SAFETY ANALYSIS REPORT (FSAR) AMENDMENT 80 (TAC NOS. M88644 AND M88645)

This letter responds to the NRC request for additional information (RAI) dated May 12, 1994. The RAI identified three issues concerning changes that were made in FSAR Amendment 80, which updated the description of various WBN accident analyses in FSAR Chapter 15. The enclosure restates these three issues and gives TVA's response to each one. Much of the information in these responses has already been discussed with Mr. Christopher Jackson of the NRC staff during telephone conversations on May 5, 1994, May 10, 1994, and May 13, 1994.

If you have any questions about the information provided in this letter, please telephone John Vorees at (615) 365-8819.

Sincerely,

Dwight E. Nunn
Vice President
New Plant Completion
Watts Bar Nuclear Plant

Enclosure
cc: See Page 2

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cc (Enclosure):

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ENCLOSURE

REQUEST FOR ADDITIONAL INFORMATION
WATTS BAR NUCLEAR PLANT
FSAR AMENDMENT 80
CHAPTER 15, ACCIDENT ANALYSIS

NRC ISSUE 1:

Please provide the basis for the use of the reactivity insertion rate of 0.6 pcm/sec in Section 15.2.4 on Page 15.2-17 of the uncontrolled boron dilution accident and state why it is changed from the previously used value of 2.0 pcm/sec.

TVA RESPONSE:

A reactivity insertion rate of 0.6 pcm/sec was used in WBN's most recent analysis of the accident transient associated with uncontrolled boron dilution during full-power operation with reactor control (i.e., control rods) in manual. This insertion rate replaced the previous value of 2.0 pcm/sec because it results in a lower (more limiting) value for the departure from nucleate boiling ratio (DNBR). DNBR is the principal parameter that is calculated in the analysis to measure the thermal-hydraulic effect of the accident on the reactor core.

A reactivity insertion rate of 0.6 pcm/sec causes slower changes to reactor power and temperature during the transient than would be the case for an insertion rate of 2.0 pcm/sec. Consequently, it takes longer to reach the reactor trip setpoint for overtemperature ΔT (OTΔT). Based on information provided by Westinghouse Electric Corporation, the OTΔT reactor trip does not occur for approximately 19 minutes. DNBR decreases throughout the transient until the reactor trip occurs. Although DNBR does not go below its limiting value for any accident scenario involving uncontrolled boron dilution, it comes closest to this limiting value for a reactivity insertion rate of 0.6 pcm/sec.

NRC ISSUE 2:

Please provide the basis for the values used for the flows to each of the four steam generators (SGs) in Cases 1b and 2b of Section 15.2.10 for the excessive heat removal due to feedwater system malfunctions accident (Pages 15.2-29 and 15.2-30).

TVA RESPONSE:

FSAR Section 15.2.10.2 defines Case 1b as accidental opening of all feedwater control valves with the reactor at zero load. Case 2b is accidental opening of all feedwater control valves with the reactor at full power. These definitions are intended to be illustrative generalizations of the actual plant conditions and valve malfunctions that were considered in these two cases.

At zero load for Case 1b, SG level is maintained with the feedwater bypass control valves. The main feedwater (MFW) control valves are closed. Each SG has a feedwater bypass control valve located in a 6" line that connects to the SG above the tube bundle area (as compared to the 16" MFW line containing the MFW control valve that connects to the SG near the bottom of the tube bundle). This bypass line is the normal flow path to the SG during plant startup and shutdown for low-load conditions. During startup, the bypass control valves are used up to approximately 22% of nominal feedwater flow. (Note: Nominal flow is the flow to each SG that is needed for steady-state operation at full-power conditions.) During shutdown, the bypass control valves are again used when feedwater flow requirements decrease to approximately 14% or less. Also during startup and shutdown, feedwater flow is limited because only one MFW pump is in operation. Normally, the motor-driven standby MFW pump (with a capacity of approximately 15% of the nominal feedwater flow to all four SGs) is used at low-load conditions. However, one of the two turbine-driven MFW pumps (each with a capacity of approximately 67% of the nominal feedwater flow to all four SGs) could also be used if the standby MFW pump was not available.

The accident scenario analyzed in Case 1b for excessive heat removal due to a feedwater system malfunction assumes that all four feedwater bypass control valves are partially open when one MFW control valve malfunctions and opens fully. It also assumes that both turbine-driven MFW pumps are in operation. This set of assumptions is very conservative since, at zero load, there is only minimal feedwater demand to dissipate reactor coolant pump heat. The feedwater bypass control valves are normally at or near their closed position. With the assumed combination of open feedwater control valves after the malfunction, the feedwater flow to each of the four SGs is 89.5%, 11.2%, 11.1%, and 12.0% of its nominal full-power value. Most of the feedwater flow is diverted to the SG which is supplied by a partially open bypass control valve and the MFW control valve that was assumed to malfunction. The small variations in flow to the other three SGs that are supplied only by partially open feedwater bypass control valves are due to differences in the flow resistance of the feedwater lines to the SGs (i.e., piping length, number of elbows, etc.).

At full power for Case 2b, SG level is maintained with the MFW control valves. The feedwater bypass control valves are closed. Both turbine-driven MFW pumps are in operation. When the feedwater system malfunction occurs in this scenario, all four MFW control valves open fully. One bypass control valve is also assumed to malfunction and open fully. The feedwater flow to each of the four SGs is 172%, 154%, 154%, and 157% of its nominal full-power value. In this case, one of the SGs receives more flow than the other three SGs because its bypass control valve was assumed to open along with its MFW control valve. The probability of such a double malfunction is negligible, but it provides a conservative basis for analysis since it creates an asymmetric flow distribution that imposes a more limiting thermal-hydraulic transient on the reactor core. As before, minor variations in flow to the other three SGs are due primarily to physical differences in piping layout.

NRC ISSUE 3:

For the large-break loss-of-coolant accident (LBLOCA) analysis, Table 15.1-2 indicates that the computer code FROTH is used. The staff finds no reference to FROTH in either the TVA submittal or FSAR Section 15.4.1. Please describe what the FROTH code is and how it is used in the analysis. Table 15.4-23 of the FSAR lists new "Minimum Safeguards ECCS Flow" for the LBLOCA. Why are these values different from those on Table 8.2 from WCAP-11627 in TVA's March 8, 1993, submittal?

TVA RESPONSE:

The FROTH computer code is used to model froth boiling in the SG tubes which can occur during a portion of the LBLOCA transient. Froth boiling affects the mass and energy releases during the LBLOCA event and, therefore, must be considered in WBN's analysis of containment design adequacy. FSAR Section 6.2.1.3.6 discusses froth boiling after the core is quenched (i.e., reflooded by the emergency core cooling system (ECCS)) for a LBLOCA transient that results from a piping break at the reactor coolant pump suction. The FSAR discussion refers to WCAP-8312-A ("Westinghouse Mass and Energy Release Data for Containment Design") for a description of the calculational methodology that was used. This Westinghouse topical report is the non-proprietary version of WCAP-8264-P-A (same title). Both WCAP-8264-P-A, Revision 1, dated August 1975, and WCAP-8312-A, Revision 2, dated August 1975, refer to the FROTH code and its use in LBLOCA analysis. Westinghouse has also informed TVA that WCAP-10325-P-A ("Westinghouse LOCA Mass and Energy Release Model for Containment Design - March 1979 Version"), dated May 1983, provides additional details about the FROTH code. Note that all of the previously mentioned Westinghouse topical reports (i.e., WCAPs) have been reviewed and approved by the NRC staff.

The ECCS flows listed in FSAR Table 15.4-23 are the correct values that were used in WBN's LBLOCA analysis for the minimum safeguards condition. Table 8.2 of WCAP-11627 ("Upflow Conversion Safety Evaluation Report - Watts Bar Units 1 & 2") also lists ECCS flows that were used for WBN's LBLOCA analysis. However, Table 8.2 does not provide a complete listing of the total ECCS flow from all ECCS pumps. The flow values that are listed in Table 8.2 are the sum of the flows from the safety injection (SI) pumps and the residual heat removal (RHR) pumps. The flow from the centrifugal charging pumps is not included, even though a note at the bottom of Table 8.2 states that it is. Westinghouse recently notified TVA by telephone that the note is in error. Also, there is a similar error in Section 8.1.4 of WCAP-11627 which incorrectly states that the centrifugal charging pump flow is included in the tabulated flow values of Table 8.2. The flow values that are actually listed in Table 8.2 can be obtained (allowing for a small round-off error) by subtracting the charging pump flow values in FSAR Table 15.4-23 from the total flow values in Table 15.4-23. The input data format for ECCS pump flow in the computer code used by Westinghouse to perform LBLOCA analysis requires that charging pump flow be entered separately from SI pump flow and RHR pump flow, which are entered as a combined value. This separation of computer code input data was the likely cause of the incomplete listing of ECCS pump flows in Table 8.2 of WCAP-11627. However, Westinghouse has confirmed that the correct ECCS pump flows, including centrifugal charging pump flow, were used in WBN's LBLOCA analysis.