UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

ENCLOSURE

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO THE SMALL-BREAK LOCA ANALYSIS DESCRIBED IN

THE FINAL SAFETY ANALYSIS REPORT FOR

WATTS BAR NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-390 AND 50-391

1.0 INTRODUCTION

By letter dated January 9, 1993, Tennessee Valley Authority (TVA) proposed to amend the Watts Bar (WBN) Final Safety Analysis Report (FSAR) to reflect a reanalysis of the small-break loss-of-coolant accident analysis (SBLOCA). The analysis of record peak cladding temperature (PCT), as reported in the FSAR by Amendment 63, was 1446.1°F. TVA committed to perform the reanalysis in a letter dated February 26, 1992. The SBLOCA reanalysis was necessary as a result of changes to the emergency core cooling system (ECCS) model that resulted in a change to the PCT of more than 50°F for the limiting accident transient. Also included in the reanalysis were reductions in the auxiliary feedwater (AFW) flow rates to represent the split flow path, and reductions in the safety injection (SI) flow rates to provide increased margin in the surveillance testing. TVA also incorporated the changes, already delineated in the FSAR and being reviewed by the staff in a separate action, pertaining to plant modifications from downflow barrel/baffle configuration to upflow barrel/baffle configuration.

2.0 EVALUATION

Westinghouse Electric Corporation performed the SBLOCA reanalysis (4-inch and 3-inch breaks) for WBN using its currently approved emergency core cooling system (ECCS) evaluation model, updated to include the changes and corrections outlined previously in TVA's letter dated July 22, 1991. The changes affecting the SBLOCA analysis are (1) fuel rod model consistency with the fuel design codes that are used to establish initial conditions for LOCA analysis; (2) the SBLOCA rod internal pressure initial condition assumption; and (3) the NOTRUMP computer code solution convergence reliability for SBLOCA analysis.

In addition to these changes, the SBLOCA reanalysis also includes revised AFW flow rates to the steam generators, and more conservative (lower) SI flow rates. TVA indicated that the revised AFW flow rates more accurately represent the split flow path from the discharge of the two motor-driven AFW pumps and the one turbine-driven AFW pump. TVA reduced the SI flow rates in the reanalysis to provide increased margin and flexibility for plant operation, maintenance, and surveillance activities.

9305200114 930517
PDR ADOCK 05000390
A PDR

2.1 Large-Break LOCA

These changes did not affect the PCT for the large-break LOCA (LBLOCA) analysis. The current LBLOCA PCT of 2126°F (see FSAR Page 15.4-4, revised by Amendment 63) was analyzed with the upflow configuration and remains more limiting, bounding the new SBLOCA analysis. By letter dated March 17, 1993, TVA estimated that the PCT will increase to 2129°F.

2.2 Small-Break LOCA Analysis

The SBLOCA reanalysis was necessary as a requirement of 10 CFR 50.46, since the cumulative changes to the ECCS model resulted in a change to the PCT of more than 50°F for the limiting accident transient. The SBLOCA analysis was performed using the approved Westinghouse ECCS Small-Break Evaluation Model and NOTRUMP, an approved digital computer code developed to determine the RCS response to design basis SBLOCA accidents and to address the NRC concerns expressed in NUREG-0611, "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse Designed Operating Reactors".

Also incorporated in this reanalysis was a revision to the AFW and SI flow rates. The initial reactor vessel internals design was analyzed for a downflow barrel/baffle region. The SBLOCA was reanalyzed for the current design, upflow barrel/baffle configuration, and includes the reduced auxiliary feedwater flow of 1050 gpm, and the reduced SI flows depicted in a draft FSAR Figure 15.3-8n. These inputs were used to analyze the 4-inch and 3-inch break sizes.

TVA determined the 3-inch break as most limiting with a peak cladding temperature of 2089°F. This is an increase of 643°F from the previous analysis of record PCT of 1446.1°F. Of the increase, 130°F is attributed to the ECCS model changes and the remaining 513°F of the increase is attributed to the combined effects of the modified AFW and SI flows.

3.0 CONCLUSION

TVA has reanalyzed its SBLOCA analysis for WBN Units 1 and 2 as required by 10 CFR 50.46, thus satisfying the commitment that was made by letter dated February 26, 1992. The reanalysis accounted for changes that were made previously to the ECCS model and also included the effects of reduced AFW and SI flows. Based on its reanalysis, TVA determined that the new PCT for SBLOCA should be 2089°F, an increase of 643°F over the previous value.

The staff finds TVA's analysis to be acceptable in that (1) approved methods were used for the analysis, (2) the PCT of 2089°F (the new analysis of record) is below the acceptance criteria established by 10 CFR 50.46 of 2200°F. The large-break LOCA still remains the most limiting event, with a PCT of 2126°F.

Principal Contributor: Sarita B. Brewer

Dated: May 17, 1993

REFERENCES

1. NUREG-0847, "SER Related to the Operation of Watts Bar Nuclear Plant, Units 1 and 2", dated June 1982.

2. Letter from W. J. Museler, Watts Bar Nuclear Plant, to USNRC, "Proposed Changes to FSAR for Reanalysis of a SBLOCA," dated January 9, 1993.

3. Letter from John H. Garrity, Watts Bar Nuclear Plant, to USNRC, "Emergency Core Cooling System (ECCS) Evaluation Model Changes (TAC No. M81154)," dated February 26, 1992.

4. Letter from E. G. Wallace, Tennessee Valley Authority, to USNRC, "Watts Bar Nuclear Plant (WBN) - Emergency Core Cooling System (ECCS) Evaluation Model Changes," dated July 22, 1991.

5. Letter from W. J. Museler, Tennessee Valley Authority, to USNRC, " Watts Bar Nuclear Plant ECCS Evaluation Model Changes", dated March 17, 1993.