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

SAFETY EVALUATION SUPPLEMENT
BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO REVISIONS TO RADIOLOGICAL CONSEQUENCES ANALYSES
RESULTING FROM POWER UPRATES
TENNESSEE VALLEY AUTHORITY
BROWNS FERRY UNITS 2 AND 3
DOCKET NOS. 50-260 AND 50-296

1.0 INTRODUCTION

On September 8, 1998, license conditions were imposed on Browns Ferry Units 2 and 3, by Amendment Nos. 254 and 214, respectively, which authorized increases in rated power. These licensee conditions state:

> TVA will perform an analysis of the design basis loss-of-coolant accident to confirm compliance with General Design Criterion (GDC)-19, and offsite limits considering main steam isolation valve leakage and emergency core cooling system leakage. The results of this analysis will be submitted to the NRC for its review and approval by March 31, 1999. Following NRC approval, any required modifications will be implemented during the refueling outages scheduled for Spring 2000 for Unit 3 and spring 2001 for Unit 2. TVA will maintain the ability to monitor radiological conditions during emergencies and administer potassium-iodide to the control room operators to maintain doses within GDC-19 guidelines. This ability will be maintained until the required modifications, if any, are complete."

The staff imposed the above condition on the power uprate because the facilities' licensing basis radiological dose models did not then treat main steam isolation valve leakage as containment bypass leakage and did not encompass consideration of the potential dose contribution emergency core cooling system leakage from piping outside containment. These considerations are consistent with Current Standard Review Plan (SRP) methodology and considered necessary to confirm that offsite doses are within Part 100 limits and control room doses are within GDC-19 limits. (At the time of licensing of the Browns Ferry facilities, the SRP was not yet available to the staff and radiological dose calculations were evaluated on a case-by-case basis.)

This Safety Evaluation supplements the staff's Safety Evaluation of August 11, 1998 which addressed control room emergency ventilation system issues raised during the course of the power uprate review.

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2.0 DISCUSSION AND EVALUATION

2.1 Licensee's Analytical Methodology

TVA's new dose calculations utilized the methodology and assumptions described in a letter dated May 1, 1998, with the exception of the following changes:

(A) The X/Q values were calculated in accordance with ARCON96.

(B) Simultaneous contamination of both control room ventilation intakes was not considered credible based on meteorological conditions and the actual physical location of the two control room intakes.

(C) Emergency Core Cooling System leakage was considered in accordance with the guidance of SRP 15.6.5, Appendix B, "Radiological Consequences of a Design Basis Loss-of-Coolant-Accident: Leakage from Engineered Safety Feature Components Outside Containment." A 5-gallon per minute leak rate into the reactor building was used.

(D) The source term inventory was updated to include a 5% power uprate and the end of life inventory for 1400 Effective Full Power Days (24 month fuel cycle), as determined by the ORIGEN computer code.

(E) Main Steam Isolation Valve (MSIV) leakage effects were calculated in accordance with NEDC-31858P, "BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems." The Technical Specification limit leak rate of 11.5 standard cubic feet per hour per valve was used.

As required by NEDC-31858P, the analysis for MSIV leakage assumed seismically rugged main steam lines which provides a path for MSIV leakage to the condenser. Modifications to the main steam lines will be required to validate this assumption.

The licensee's methodology conforms to relevant staff positions.

2.2 Results of Reanalysis

The analysis results below indicate the following doses in the Main Control Rooms, the exclusion area boundary and the low population zone in roentgen equivalent man (rem):

	CONTROL ROOM	SITE BOUNDARY	LOW POPULATION ZONE
THYROID	4.059	5.837	50.92
GAMMA	0.6716	0.1664	0.3461
BETA	0.04753	0.1005	0.3502

These doses are well within the acceptance criteria prescribed by 10 CFR 50 Appendix A, GDC-19 (control room dose) and 10 CFR 100 (offsite dose).

2.3 Acceptability of Licensee's Methodology and Results

The licensee's dose models conform to accepted methods for calculating the radiological consequences of design basis accidents as stated in the SRP. The calculated doses are within the acceptance criteria of Part 100 and GDC-19.

3.0 CONCLUSION

The licensee has satisfactorily met the license condition requirement to perform an analysis and submit the results to the NRC for review and approval.

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Date: August 3, 1999