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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 70 TO FACILITY OPERATING LICENSE NO. DPR-68

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNIT NO. 3

DOCKET NO. 50-296

1.0 Introduction

By letter dated January 23, 1984 (TVA BFNP TS 195), the Tennessee Valley Authority (the licensee or TVA) requested changes to the Technical Specifications (Appendix A) appended to Facility Operating License No. DPR-68 for the Browns Ferry Nuclear Plant, Unit 3. The proposed amendments and revised Technical Specifications were to: 1) incorporate the new physics and thermal-hydraulic limits associated with the sixth fuel cycle, and 2) reflect modifications performed during the refueling outage. The amendment addresses the changes to the Technical Specifications associated with the core reload and the thermal power monitor; the other changes associated with the modifications will be addressed by a separate evaluation.

2.0 Discussion and Evaluation

In support of the Cycle 6 reload, TVA submitted with its January 23, 1984 application a Reload Licensing Report (TVA-RLR-001) which describes results of the core design and safety analyses performed for Cycle 6. This reload is the first to be analyzed by TVA instead of the fuel vendor.

The Cycle 6 core will consist of 248 fresh fuel assemblies, and 516 burned assemblies that were originally loaded in Cycles 2 through 5. Among the burned assemblies are eight lead test assemblies that were initially installed and approved for Cycle 5. The remaining fuel is of the standard GE design described in NEDE-24011-P-A(US) GESTAR II "General Electric Standard Application for Reactor Fuel" January 1982.

Nuclear Design

The shutdown margin was determined by using the TVA BWR simulator code to calculate the core multiplication at selected exposure points for Cycle 6, with the strongest rod fully withdrawn. The shutdown margin was calculated to be 1% at the point in the cycle at which it is minimum. This exceeds the Technical Specification requirement of 0.38% and is, therefore, acceptable.

The Standby Liquid Control System (SLCS) is designed to provide the capability of bringing the reactor subcritical at any time in a cycle, from a full power, xenon free, condition to a cold, all rods out condition. The SLCS shutdown margin is calculated using the BWR simulator code to be 0.019 delta k with a 600 ppm boron concentration.

Reactivity coefficients are not used in the TVA analyses; however, their values are generated and reported. The void coefficient is calculated to be -0.757% delta k/% void at 100% flow and -0.744% delta k/% void at 105% flow. These values are consistent with those customarily obtained for BWR reloads and are acceptable.

Thermal Hydraulics

The safety limit minimum critical power ratio (SLMCPR) of 1.07 is based on the GEXL correlation previously used for BF-3. When meeting this SLMCPR during a transient, at least 99.9% of the fuel rods in the core are expected to avoid boiling transition.

Various transient events can reduce MCPR from its normal operating level. To assure that the SLMCPR will not be violated during abnormal transients, the most limiting transients have been reanalyzed for this reload. The events analyzed were load rejection without bypass, feedwater controller failure, loss of feedwater heaters, fuel loading errors, and control rod withdrawal errors. The anticipated transients are analyzed to determine that which yields the larges reduction in CPR and that value is added to the safety limit value to obtain the operating limit MCPR.

Core wide pressurization transients have been analyzed by TVA with the TVA-RETRAN code. This code has been described in a topical report (TVA-TR81-01, "BWR Transient Analysis Model Utilizing the RETRAN Program", TVA, December 1981) which also includes the verification of the code. This report was reviewed and approved by the staff with two possible restrictions by our letter to TVA of April 7, 1983. To remove these conditions, TVA submitted by letter dated November 21, 1983 a report, "Validation fo COMETHE III-J for Gap Conductance Calculations". Based on our review, our letter of May 23, 1984 advised TVA that TVA TR 81-01 was approved without conditions for referencing by TVA in core reload analyses performed by TVA for BWR facilities operated by TVA.

The non-pressurization events were analyzed with the TVA three dimensional core simulator code (TVA-TR78-03A, "Three-Dimensional Core Simulation Methods", TVA, January, 1979) which we approved by our letter to TVA of October 16, 1979. These potential transients are either steady state events or very slow transients.

The calculated MCPR's necessary to prevent SLMCPR violation during each transient are presented in the Reload Licensing Report (RLR). The limiting events for establishing the OLMCPR are the load rejection without bypass event (pressurization) and the rod withdrawal error (non-pressurization). When the reactor is operated in accordance with the proposed OLMCPR, the SLMCPR will not be violated in event of an abnormal operating transient. Changes to the Technical Specifications will incorporate the new OLMCPR.

A curve of MCPR as a function of average scram insertion time has been updated for the Technical Specifications.

Operation at 105 Percent Rated Flow

TVA proposed to operate at flow rates up to 105 percent of rated flow during Cycle 6. Analyses have been performed at both 100 and 105 percent flow and the more limiting results used to establish operating limits. The flow-biased instrumentation for the rod block monitor will be signal clipped for a setpoint of 106 percent since flow rates higher than rated would result in a delta CPR higher than reported for the rod withdrawal error event.

Such operation has been previously approved for Cycle 5 and continues to be acceptable for Cycle 6.

Loss of Coolant Accident (LOCA)

TVA submitted an addenda to the "Loss of Coolant Accident Analysis for Browns Ferry Nuclear Plant Unit 3" prepared by the General Electric Company (GE) (NEDO-24194A), with the Cycle 6 reload application. The addenda covers the new BP8DRB284L fuel assemblies (FAs). The maximum average Planar Linear Heat Generation Rate (MAPLHGR) versus Planar Average Exposure for the most limiting break size were calculated by General Electric using the CHASTE code.

The CHASTE code is used, with inputs from other codes, to calculate the fuel cladding heatup rate, peak cladding temperature (PCT), peak local cladding oxidation, and core-wide metal-water reaction for large breaks. The detailed fuel model in CHASTE considers gap conductance, clad swelling and rupture, and metal water reaction. The empirical core spray heat transfer and channel melting correlations are built into CHASTE, which solves the heat transfer equations for the entire LOCA transient at a single axial plane in a single FA. Iterative applications of CHASTE determine the maximum permissible planar power where required to satisfy 10 CFR 50.46 acceptance criteria for emergency core cooling.

The MAPLHGR values and peak cladding temperatures for each type FA that will be in the BF-3 Cycle 6 reload are presented in NEDO-21494A (as addended). The limit MAPLHGR values for the new BP8DRB284L fuel are included as proposed Technical Specifications changes in TVA's submittal, the values for other type FAs having been previously included. These MAPLGHR values will, in event of a LOCA, limit PCT to less than that allowed by 10 CFR 50 Appendix K and are, therefore, acceptable.

Control Rod Drop Accident

The rod drop accident (RDA) was reanalyzed for Cycle 6 by TVA using the TVA RDA transient simulation program with input from the TVA 3D simulator code. The RDA simulation model is described in Appendix A of the RLR. The TVA code has been checked against a test problem using a method similar to that of the fuel vendor and shown to be conservative. The staff therefore concludes that the RDA analysis method used by TVA is acceptable. The results of the analysis for BF-3, Cycle 6 is 240 cal/gram maximum fuel enthalpy. This value meets the staff acceptance criterion of 280 cal/gram and is therefore acceptable.

Overpressure Analysis

The licensee has reanalyzed the limiting pressurization event - main steamline isolation valve (MSIV) closure followed by direct neutron flux scram, using the TVA-RETRAN code. The results indicate a peak vessel pressure of 1287.6 psia. This is substantially identical to that of Cycle 5, reported as 1272 psig in our March 29, 1982 Cycle 5 evaluation and found acceptable therein.

Thermal-Hydraulic Stability

A thermal-hydraulic stability analysis was performed for Cycle 6 using a model based on the LAPUR code which is applicable to both core and channel hydrodynamic stability. This model is currently being reviewed by the staff. The review has not progressed to the point where the staff can give generic approval to the TVA methodology. However, the review has progressed sufficiently to approve the Cycle 6 reload for the following reasons:

1. There are no significant changes in fuel loading between Cycle 6 and Cycle 5.

2. The decay ratio (core) as calculated by TVA for Cycle 6 is 0.87 which is very similar to the Cycle 5 calculated decay ratio and is acceptable.

3. The TVA model adequately predicts the results of the Peach Bottom Thermal-Hydraulic Stability Tests.

Thermal Power Monitor

The APRM flow-biased flux trip will be altered by the insertion of a damping circuit having a six-second time constant. This circuit simulates the time constant for heat transfer from fuel to coolant such that the flow-biased trip is based on heat flux as opposed to neutron flux. The fixed trip will still respond directly to neutron flux.

The Technical Specifications will be revised to reflect the modification.

The thermal power monitor has been previously approved for use on other BWRs including Browns Ferry Units 1 and 2 (i.e. BF-2 Amendment 85, BF-1, Amendment 91). The staff therefore concludes that it is acceptable for BF-3.

3.0 <u>Changes to Technical Specifications-Reload</u>

Specification 3.5.I and the Table of Contents will be changed to include the "MAPLHGR vs AVERAGE PLANAR EXPOSURE" table for the new BP8DRB284L type fuel.

Specification 3.5.K will be changed to update MCPR limits for Cycle 6. The Table of Contents will be revised to reflect the new page number of Figure 3.5.K-1.

Bases for Limiting Safety System Settings Related to Fuel Cladding Integrity, and Reactor Coolant System Integrity will be revised to reflect that reload analyses are being done by TVA instead of GE. Changes in text and references reflect TVA methodology.

The staff has reviewed these changes and concludes they are acceptable. This conclusion is based on the following:

1. Approved methods were used to perform the design and analysis of the Cycle 6 reload or the approval could be granted on other grounds.

2. Appropriate criteria for operational limits and accident consequences were met.

4.0 <u>Changes to Technical Specifications - Thermal Power Monitor</u>

Technical Specifications Sections 2.1.A Fuel Cladding Integrity Limiting Safety System Settings and Bases), 3.1 (Reactor Protection System, Limiting Conditions for Operation) and 4.1 (Reactor Protection System Surveillance Requirements) will be changed to reflect the addition of the thermal power monitors. The staff has reviewed the changes and found them to be acceptable. The changes are consistent with those issued for Unit 2 in Amendment 85 of the Unit 2 Technical Specifications.

5.0 <u>Environmental Considerations</u>

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupation radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for catergorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

4.0 Conclusion

We have concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

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