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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 78 TO FACILITY OPERATING LICENSE NO. DPR-68

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNIT 3

DOCKET NO. 50-296

1.0 INTRODUCTION

By letters dated January 23, 1984 and June 6, 1984 (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 amendment and revised Technical Specifications would (1) incorporate the limiting conditions for operation of the facility in the sixth fuel cycle following the fifth refueling of the reactor and (2) reflect modifications performed during the current cycle 5 outage. This amendment addresses the changes to the Technical Specifications associated with all of the various modifications completed during this refueling outage except for the installation of a reactor protection system power monitoring design modification. The latter modification will be addressed separately. The core reload was addressed in a separate amendment (Amendment No. 70).

2.0 DISCUSSION

The modifications and the changes to the Technical Specifications were described in the Commission's Notice of this application published pursuant to Public Law 97-415 on May 23, 1984 (49 FR 21841).

3.0 EVALUATION

3.1 Changes Related to Torus Modifications

One of the changes to the Technical Specifications (TS) is to revise the tables that list the surveillance instrumentation associated with the suppression pool bulk temperature. This modification provides an improved torus temperature monitoring system which consists of 16 sensors. This will provide a more accurate indication of the torus water bulk temperature as required by NUREG-0661 and will replace the suppression chamber water temperature instruments presently listed in the TS. The proposed changes to the TS impose operability and calibration requirements on the new temperature monitoring system and delete these requirements for the old system.

Another change to the TS is to revise the bases for the present limits on temperature of water in the torus. The present bases for suppression pool temperature limits were founded on the Humboldt Bay and Bodega Bay tests. Consistent with the long-term torus integrity program of NUREG-0661 and NUREG-0783, the bases require change to account for steam mass fluxes through the safety/relief valve (S/RV) T-quenchers. The proposed bases describe assurances of stable and complete condensation of steam discharged through the S/RVs and adequate residual heat removal (RHR) and core spray pump net positive suction head. The bases do not contain any limits or action requirements; they provide the justification for the limiting conditions of operation and the surveillance requirements.

Section 4.5.B.1 of the Technical Specifications requires that every three months, the LPCI capability of the RHR pumps be demonstrated. In the tests, the pumps take suction from the torus and return the water to the torus. The pumps are required to demonstrate that two pumps in the same loop can deliver at least 15,000 gpm against an indicated system pressure (head) of 200 psig.

The two-pump 15,000 gpm LPCI surveillance test was found to induce vibrations in the RHR return line to the torus. To eliminate the vibration, an orifice has been installed in the return line. However, installation of this orifice plate also decreases the suppression pool cooling mode of RHR operation from 15,000 gpm to approximately 12,000 gpm. A new containment cooling analysis was performed for this configuration, and it was determined that this flow rate induces a long-term suppression pool temperature well within that necessary for stable and complete steam condensation and for adequate RHR and core spray pumps net positive suction head. The revised test requirement is that the two pumps demonstrate that they can deliver 12,000 gpm against a higher head of 250 psig. The orifice is in the return line to the torus and does not change the volume of water that would be injected into the reactor during the LPCI mode. The 12,000 gpm at higher pump head pressure is equivalent to 15,000 gpm at lower discharge pressure. We conclude that the change has no adverse impact on the LPCI or containment cooling modes of RHR operation.

Section 4.7.A.2.k of the present Technical Specifications requires that if extended relief valve operation causes the temperature of the suppression pool to exceed 130°F, the reactor shall be shutdown and the torus and drywell visually inspected for signs of distress or displacement. Since the torus is being extensively upgraded to withstand dynamic loading significantly beyond that originally expected, extended operation of relief valves above a suppression pool temperature of 130°F is not expected to be a safety concern warranting placing the reactor in cold shutdown and performing a torus inspection. Therefore, this requirement is being deleted.

The Technical Specifications "Bases" for primary containment contain specific references to drywell and suppression chamber coatings. There is some variation between the Browns Ferry units in the type and application of the coating, particularly due to the Mark I modification program;

therefore, the Technical Specification bases are being generalized so that Technical Specification changes will not be required each cycle.

The changes described above were approved for Unit 1 in Amendment No. 92, dated December 12, 1983, for Unit 2 in Amendment No. 85, dated March 11, 1983 and are acceptable for Unit 3.

3.2 Scram Discharge Instrument Volume

The scram discharge volumes (SDVs) and SDIVs are being modified to address inadequacies identified by the partial rod insertion event on Browns Ferry Unit 3 in June 1980. One of the modifications includes adding another valve in series to the existing drain and vent valves on the SDV and SDIV. Another modification includes adding electronic level switches to initiate a scram on a high level in the SDIV. On June 24, 1983, the Commission issued Orders for the Browns Ferry Nuclear Plant, Units 1 and 3 to install permanent Scram Discharge System modifications during the Cycle 5 outages for Units 1 and 3 in accordance with Generic Letter 81-09, BWR Scram Discharge System. The modifications have been previously completed for Units 1 and 2. Both the modification of the systems and submission of TS changes to place operability and surveillance requirements on the new instruments and valves were required of the licensee to be in compliance with the Commission Order.

We have reviewed the proposed changes to the TS and conclude that the proposed changes are consistent with the staff guidelines stated in the December 1, 1980 BWR Scram Discharge System Safety Evaluation. Further, these same proposed changes have been previously approved for Browns Ferry Unit 1 by Amendment No. 92 and for Browns Ferry Unit 2 by Amendment No. 85. Thus, we conclude that the proposed changes in the TS for Unit 3 are acceptable.

3.3 Accident Monitor Instrumentation

Item II.F.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," requires all licensees to install six new monitoring systems to provide improved accident monitoring capability. For all six categories, NUREG-0737 states, "Changes to Technical Specifications will be required." During this refueling outage, the licensee has installed: a drywell wide-range pressure monitoring system and a suppression chamber wide-range water level monitoring system. These items were required by NUREG-0737, items II.F.1.4 and II.F.1.5, respectively. The changes to the TS, which track the model TS provided to the licensee by the staff, are to add operability and surveillance requirements on the new monitoring systems and are acceptable.

3.4 Scram Permissive Pressure Switches

A number of scram bypasses are provided in the RPS system to account for the varying protection requirements depending on reactor conditions and to allow for instrument service during reactor operations. Some bypasses are automatic, others are manual. At Browns Ferry and most other BWRs, whenever the mode switch is in the shutdown, refuel or startup/hot standby position - i.e., in any position other than RUN - there is an automatic bypass of the scram trips from the main steamline isolation scram and main condenser low vacuum scram if the reactor pressure is below 1055 psig. The bypass allows reactor operations at low power with the main steamlines isolated and the main condenser not in operation. These conditions exist during startups, certain reactivity tests during refueling, and hot standby conditions. This is commonly referred to as "bottled-up operation." In early 1974, as part of the startup test program for Browns Ferry Unit 1, the General Electric Company (GE) conducted a test of the BWR-4 product line which demonstrated that there was no stability problem with bottled-up operation up to full pressure and temperature conditions. Rather than bypass the main steamline isolation valve closure and the turbine condenser low vacuum scram functions whenever the reactor pressure is below 1055 psig, the licensee proposes to delete this scram system except when in the RUN mode. The reactor high pressure scram is set at 1055 psig and is operable in the refuel and startup/standby modes of operation. If reactor pressure exceeds 1055 psig, the reactor scrams due to the reactor high-pressure scram function, and the main steamline isolation valve closure and the turbine condenser low vacuum functions become operable. The bypass circuit therefore serves no real purpose. When the two scram functions become available, the reactor is already scrammed.

As noted above, the two scram functions which the licensee proposes to delete in the startup, shutdown and refuel modes are automatically bypassed in these modes, unless the reactor pressure should exceed 1055 psig - and, in this case, the system is tripped by the high pressure scram. A review of the original basis for the automatic bypass justification indicates that the proposed change is not likely to affect the probability or consequences of previously-analyzed accidents one way or the other. Since the core is protected by a high pressure trip at 1055 psig in all these modes, these two scram functions serve no useful purpose in the refuel and startup/hot standby modes.

Although no change has been proposed regarding "bottled-up" operation (operation with MSIVs closed), TVA submitted a report presenting the results of calculations and tests which show such operation to be safe. We find this acceptable and also conclude that the requested TS changes are acceptable.

3.5 H_2/O_2 Analyzer Isolation Valves

In our Safety Evaluation for Amendment No. 37 (January 12, 1981) we approved TVA's plans for replacement of the old analyzer systems with the new Hay-Republic systems. The approved changes were based on TVA's plans, as stated in its letter dated September [illegible], 1980, to provide an inboard and an outboard isolation valve for each sample line. In the January 23, 1984 submittal TVA states that the inboard valves have been moved outboard thus giving two outboard, and no inboard isolation valves. TVA thus has requested TS amendments to the isolation valve table to reflect the modifications.

These changes are acceptable based on consistency with Regulatory Guide 1.11 staff positions on instrument line isolation valves.

3.6 Isolation Valve for Torus Demineralized Water Line

During the current refueling outage primary containment isolation valve 2-1143 is being removed. This valve isolates the demineralized water line to the torus ring header. The line is no longer used so the valve will be removed and the line capped. Therefore, the licensee proposes to change Table 3.7.E to delete the requirement for periodic leakage testing. This is acceptable based on the valve being replaced by a cap which is not subject to leakage.

3.7 Testable Penetrations

TS Table 3.7.B lists testable penetrations with double O-ring seals which are required by TS 4.7.2.g to be periodically leak tested. TVA has modified flanges on the following valves to make them testable and proposes to include them in the table: 64-18, 64-19, 64-20, 64-21, 64-29, 64-31, 64-32, 64-34, 76-17, 76-18, and 84-8A-D. Other changes to this table include (1) penetration X-35G presently identified as "T.I.P. Drives" is being revised to indicate that it is a "spare;" (2) penetration X-219A (spare) is being added; and (3) "DW Flange-Top Head" is being changed to "Drywell Head," and penetration X-213A is being deleted. These changes will update the table to bring testing requirements into conformance with 10 CFR 50 Appendix J for all testable penetrations with double O-ring seals, and are, therefore, acceptable.

3.8 Administrative Changes

The Table of Contents will be revised to be consistent with the titles and page numbers in the TS.

In the Bases to Section 2.1 the subsection on "IRM-Flux Scram Trip Setting" will be revised for consistency with Units 1 and 2 and FSAR 7.5.5.4. (Note: This change was inadvertently included on page 21 of Amendment No. 70.)

In Tables 3.2.B and 4.2.B "condensate storage tank level" will be changed to "condensate header level" to more accurately reflect the physical arrangement to the HPCI suction switchover instrumentation.

These changes are administrative or editorial in nature. They do not revise any safety limits, limiting safety system settings, limiting conditions for operation, bases or administrative requirements and are acceptable.

3.9 Modification Postponed

By letters dated March 27, 1984 and June 6, 1984, TVA advised us that three of the plant modifications scheduled to be completed during this current outage were being postponed. These three modifications were (1) adding a redundant air supply to the drywell, (2) replacing certain mechanical-type switches in the reactor protection system with analog loops and (3) replacing the existing containment radiation monitoring system with high-range instruments meeting the requirements of NUREG-0737, item II.F.1.3. Changes to the TS for these modifications were submitted in the licensee's application of January 23, 1984 and were described in the Commission's Notice of this application. TVA's letter of June 6, 1984 submitted revised TS pages to reflect the deletion of changes associated with these three modifications. The June 6, 1984 letter did not add any new changes that were not described in the Notice.

4.0 ENVIRONMENTAL CONSIDERATIONS

The amendment involves changes in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and in surveillance requirements. 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 occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical 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 the amendment.

5.0 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 the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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