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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 125 TO FACILITY OPERATING LICENSE NO. DPR-52

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNIT 2

DOCKET NO. 50-260

1.0 INTRODUCTION

By letter dated August 23, 1984 (TVA BFNP TS-199), as supplemented September 4 and November 13, 1984, April 3, May 8, June 27, November 20 and December 30, 1985, and April 29, 1986, the Tennessee Valley Authority (the licensee or TVA) requested an amendment to Facility Operating License No. DPR-52 for the Browns Ferry Nuclear Plant, Unit 2. The proposed amendment would change the Technical Specifications (TS) of the operating license to: (1) modify the core physics, thermal and hydraulic limits to be consistent with the reanalyses associated with replacing about one-third of the core during the Cycle 6 core reload outage and (2) reflect changes in various specifications as a result of plant modifications performed during the outage. In addition, TVA has updated the TS pages involved and made administrative corrections.

The areas involved in the amendment are as follows:

A. Core related changes

B. Changes related to torus modifications

C. Miscellaneous plant modifications

1. Reactor protection system (RPS) modification
2. Scram discharge instrument volume
3. Analog trip system
4. Scram permission pressure switches
5. Drywell temperature and pressure
6. TMI Action plan items (NUREG-0737)
7. Testable penetrations
8. Redundant air supply to the drywell
9. Demineralized water isolation valve
10. Residual heat removal (RHR) head spray

D. Administrative changes

2.0 EVALUATION

A. Core related changes

TVA made application to amend the Technical Specifications of Browns Ferry Nuclear Plant, Unit 2. The changes were required, in part, in order to permit the reloading and operation of Unit 2 for Cycle 6. In support of the application TVA submitted a Reload Licensing Report (Reference 1). The staff has reviewed this document and prepared the following evaluation of those aspects of the application pertaining to the reload.

Reload Description

For Cycle 6, 300 irradiated fuel assemblies will be removed from the core and replaced by 296 General Electric P8X8R assemblies and 4 Westinghouse designed QUAD + demonstration assemblies. In addition, the reload analysis has been performed by TVA, with the exception of the LOCA analysis which has been done by General Electric. The demonstration program has been described and analyses performed on the effect of the QUAD + assemblies on the core parameters by Westinghouse Nuclear Energy System, the manufacturer of the assemblies. TVA has submitted a report, WCAP-10507, "QUAD + Demonstration Assembly Report" (Reference 2) for the description of the program and its effects. The use of increased core flow is planned for Cycle 6. Analyses were performed for both 100 percent and 105 percent of rated flow and the most conservative results were used in determining the operating limits.

Fuel Mechanical Design

The P8X8R assemblies to be loaded into the core are identical to those inserted in Cycle 5. They are standard General Electric BWR fuel assemblies which are described in the GESTAR document (Reference 3) and we conclude that no further review of these assemblies is required. The mechanical design of the four QUAD + assemblies is described in Reference 2. That document also describes the fuel rod design analysis. The acceptability of these analyses for Lead Test Assemblies is the subject of a separate evaluation (Attached). That evaluation concludes that the QUAD + assemblies may use the various fuel rod design criteria of the P8X8R fuel on an interim basis for the Lead Test Assemblies.

Nuclear Design

This reload is the first one performed for Unit 2 by the licensee. The analysis methods used by TVA are described in References 4, 5 and 6. These reports have been reviewed and approved by the staff for use in such analyses. The results of the analyses are reported in Reference 1. The shutdown margin is calculated to be 1.0 percent reactivity change at the point in the cycle at which it is a minimum. This value exceeds the Technical Specification requirement of 0.38 percent and is acceptable. The standby Liquid Control System provides a shutdown margin of 1.8

percent reactivity change with a boron concentration of 600 ppm boron. This is an acceptable value. Reactivity coefficients are not used in the performance of transients by TVA. However, a void coefficient is obtained in the process of collapsing from 3-D to 1-D cross-sections. This value is in the range of those customarily obtained for BWR reload cores and is acceptable. The effect of the presence of the four Quad + assemblies on the neutronic behavior of the core is discussed in Reference 2, which is the subject of a separate evaluation (Attached). That evaluation concludes that the presence of the four QUAD + assemblies has a negligible effect on core neutronics. TVA has performed cycle specific analyses and concurs with the conclusions of the Westinghouse report. We conclude that the nuclear design and analysis of the Cycle 6 core are acceptable.

Thermal-Hydraulic Design

The thermal-hydraulic analysis of the Browns Ferry Unit 2 Cycle 6 reload has been reviewed to determine whether acceptable thermal-hydraulic limits have been met, whether acceptable analytical methods were used and whether the core exhibits thermal-hydraulic stability.

Safety Limit MCPR

The GEXL Critical Heat Flux Correlation is used to obtain the value of the safety limit MCPR. This correlation has been previously used for Browns Ferry Unit 2 and continues to be acceptable. The value of 1.07 for the safety limit MCPR is generic for BWR reloads and is acceptable.

Operating Limit MCPR

The procedures and techniques used to obtain the value of the operating limit MCPR are described in Reference 7 which has been reviewed and approved by the staff. The anticipated transients are analyzed to determine that which yields the largest reduction in CPR. That value is then added to the safety limit value (1.07) to obtain the operating limit MCPR. For the pressurization events both Option A and Option B limits are obtained. The results were calculated for the P8X8R fuel. The QUAD + fuel will be loaded into non-limiting core locations and monitored to the same operating MCPR limits.

Operation at 105 Percent of Rated Flow

The licensee proposes to operate at core flow rates up to 105 percent of rated flow for Cycle 6. Such operation has been approved for Cycle 5 in Browns Ferry Unit 2 and it continues to be acceptable for Cycle 6. Analysis of Cycle 6 operation has taken into account such operation.

Core Thermal-Hydraulic Stability

TVA uses a computerized model for analysis of boiling water reactor (BWR) stability for Cycle 6 of Browns Ferry Unit 2. The analysis model is based on the LAPUR computer code and is applicable to both core and channel hydrodynamic stability. It is the same model which was used for the analysis of the previously approved Browns Ferry Unit 3 Cycle 6 reload.

The model proposed by TVA has been under review by the staff. The safety evaluation of this model has not yet been issued but the review has progressed sufficiently for the staff to approve the TVA analysis of Cycle 6 of Browns Ferry Unit 2 for the following reasons.

1. The only significant change in fuel loading between Cycle 6 of Browns Ferry Unit 2 and the previously approved and currently operating Cycle 5 of Unit 2, is the addition of the four QUAD + demonstration assemblies. The stability characteristics of these assemblies were reviewed separately (see next section) and found acceptable.

2. The decay ratio as calculated by the TVA model for Cycle 6 of Browns Ferry Unit 2 is .71, which is lower than the calculated decay ratio (.73) of the previously approved Cycle 6 of Browns Ferry Unit 3.

3. The TVA model does a good job in predicting the results of the Peach Bottom Thermal-Hydraulic Stability Tests.

Presence of QUAD + Assemblies

The thermal-hydraulic performance of the QUAD + assemblies is discussed in Reference 2. The evaluation of that reference (Attached) concludes that use of QUAD + bundles as demonstration assemblies is acceptable provided that the guidelines of Section 4.1 of Reference 2 are followed and that a cycle specific analysis shows at least a margin of 20 percent in power between the QUAD + assembly and the lead assembly at full power and flow conditions. TVA has confirmed that the guidelines were followed and performed analyses to show that a 27 percent power margin exists for Cycle 6. The staff asked Westinghouse to show that the stability characteristics of the QUAD + assemblies are acceptable for inclusion in the Browns Ferry Unit 3 Cycle 6 core. The results of Westinghouse's analytical evaluation which qualifies the QUAD + stability margin is presented in Reference 2. The focus of this evaluation is on individual channel stability since the small number of QUAD + demonstration assemblies in the core will not have any significant impact on the core average parameters and hence not affect overall core stability. The Westinghouse analysis show the QUAD + assemblies to have an additional margin of 0.15 in decay ratio when compared to the P8X8R fuel already in the core. The Westinghouse evaluation used parametric analyses based on published data to quantify the relative stability margin of the QUAD + demonstration assembly compared to the P8X8R fuel and did not perform detailed stability calculations for the QUAD + assembly itself.

The staff reviewed the analysis performed by Westinghouse in Reference 2 and has found it to be a reasonable method for approximating the stability margin for the QUAD + assembly. While the staff finds that such an approach is acceptable for the limited number (4) of QUAD + assemblies in the core it is very approximate and considerably more detailed calculations would be required to justify a full reload of QUAD + assemblies. We conclude that the thermal-hydraulic design and analysis for Browns Ferry Unit 2 Cycle 6 are acceptable.

Transient and Accident Analysis

Core-wide pressurization transients were analyzed with the TVA-RETRAN (Reference 7) code which has been reviewed and approved by the staff. The two conditions cited in the review use of the COMETHE-III J code and approval of the parent RETRAN code, has been satisfied. Use of TVA-RETRAN is therefore acceptable.

The nonpressurization events were analyzed with the three dimensional core simulator code (Reference 5) since these are either steady state events or very slow transients. The limiting pressurization transient is the Load Rejection Without Bypass and the limiting nonpressurization events are the Loss of Feedwater heater and Mislocated Bundle Error. Since the replacement fuel is identical to some of the fuel already present in the core, reanalysis of the LOCA event was not required. Reference 2 presents analyses to show that the MAPLHGR limits for the P8DRB284L assemblies can be conservatively applied to the QUAD + assemblies. The rod drop accident analysis was performed with the methodology described in Reference 8. This methodology was approved for use in the Cycle 6 reload analysis for Browns Ferry Unit 3 and is acceptable for Unit 2. The result of the analysis for Cycle 6 of Browns Ferry Unit 2 is 152 calories per gram peak fuel enthalpy. This value meets our acceptance criterion of 280 calories per gram for this event and is acceptable.

Technical Specification Changes

Scram Permissive Pressure Switches at 1055 PSIG

Current Technical Specifications require the main steam line isolation valve closure and the turbine condenser low vacuum scram functions to be operable in the refuel, startup/standby, and run modes. However, these trips are bypassed in the refuel and startup/standby modes unless the reactor pressure is greater than 1055 psig. Since the core is protected by a high pressure trip at 1055 psig in all modes the two scram functions serve no useful purpose in the refuel and startup/hot standby modes. TVA proposes to delete the requirement for operability of the scram functions in those modes and to remove the bypass function. As a result of our review of this area of operation, we agree that these scram requirements accomplish no useful purpose in these modes. We conclude that the proposed Technical Specification change is acceptable.

MCPR-MAPLHGR Specifications

The operating limit MCPR as a function of average scram time, τ, has been altered to account for the Cycle 6 reload. The proposed curve (Figure 3.5.K-1) is consistent with the value given in the reload report (Reference 1) and is acceptable.

The MAPLHGR tables have been revised by deleting those for fuel types no longer present in the core and consolidating the data into two tables, 3.5.I-1 and 3.5.I-2. No changes have been made in the MAPLHGR values. The values for the P8DRB284L type are to be used for the QUAD + fuel. Such use is justified in Reference 2 for demonstration assemblies and is acceptable.

Reference in Bases

At various locations, the Technical Specification Bases have been revised to reflect the fact that the safety analyses were performed by TVA. These revisions are acceptable.

Based on the review described above, we conclude that Browns Ferry Unit 2 may be loaded and operated for Cycle 6. This includes the presence of four QUAD + bundles as lead test assemblies. This conclusion is based on the following:

1. The safety analyses have been performed by previously approved methods and procedures, except for those directly relating to the demonstration assemblies.

2. The use of the demonstration assemblies has been approved (see Attached evaluation) subject to certain conditions. These conditions have been met for Browns Ferry 2 Cycle 6.

3. The Cycle 6 core meets all the staff's acceptance criteria.

B. <u>Changes Related to Torus Modifications</u>

One of the changes to the 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. This change has been previously approved for Unit 3 by Amendment No. 78 dated August 27, 1984.

The change to the TS are necessary follow up actions essential to the implementation of this improvement. The changes to the TS place operability and calibration requirements on the new temperature monitoring system. Since these are new instruments, the surveillance requirements are not presently in the TS.

We have reviewed this proposed change and find it consistent with NRC guidance and it is, therefore, acceptable.

C. Miscellaneous plant modifications

1. Reactor Protection System (RPS) Modifications.

By letter dated August 7, 1978, the Commission advised TVA that during review of Hatch Unit 2, the staff had identified certain deficiencies in the design of the voltage regulator system of the motor generator sets which supply power to the reactor protection system (RPS). Pursuant to 10 CFR 50.54(g), TVA was required to evaluate the RPS power supply for Browns Ferry 1, 2 and 3 in light of the information set forth in our letter. By letter dated September 24, 1980, the staff informed TVA (and most other BWRs) that "we have determined that modifications should be performed to provide fully redundant Class IE protection at the interface of non-Class IE power supplies and RPS." The staff also advised TVA that "we have found that the conceptual design proposed by the General Electric Company and the installed modification on Hatch are acceptable solutions to our concern." By letter dated December 4, 1980, TVA committed to install the required modifications. By letters dated October 30, 1981 and July 28, 1982, NRC sent TVA model Technical Specifications for electric power monitoring of the RPS design and modifications.

By letter dated June 27, 1985, the staff approved the TVA proposed design modifications to the RPS power supply system. During the current outage of Unit 2, the RPS is being modified to provide a fully redundant Class IE protection at the interface of the non-Class IE power supplies and the RPS. This will ensure that failure of a non-Class IE reactor protection power supply will not cause adverse interaction to the Class IE reactor protection system.

The Technical Specifications are being revised similar to the model TS provided to TVA to reflect the limiting conditions for operation and surveillance requirements associated with the RPS modifications. Page 42 is being modified to add a description of these sections in the Bases.

Based on our Safety Evaluation dated June 27, 1985, and the TS submitted, we find the proposed amendment acceptable.

2. Scram discharge instrument volume

The scram discharge instrument volumes (SDIVs) were modified to address inadequacies identified by the partial rod insertion eventon Browns Ferry Unit No. 3 in June 1980[(1)]. The modifications of interest to this Safety Evaluation involve replacing the scram discharge tank's float devices

(1) Briefly, an undetected accumulation of water in the SDV reduced the available free volume for discharge of scram water which inhibited insertion of the control rods. The level detection system utilized float type instruments and an inspection of the instruments turned up several floats that had been damaged. It could only be concluded that the floats had been subjected to harmful hydrodynamic forces.

with new electronic level instruments. These instruments will initiate a scram on high level.

Tables 4.1.A and 4.1.B were revised to reflect changes to the required surveillance testing on the two electronic level switches. The acceptability of the changes to the surveillance testing will be addressed in Section C-3 of this SE.

Based on our review, we conclude that the proposed modifications to the Technical Specifications in the instrumentation and controls area are acceptable. The basis for our determination is that the modifications are consistent with the staff guidelines as stated in the BWR Scram Discharge Safety Evaluation Report, dated December 1, 1980. In addition, these proposed modifications have been previously approved for Browns Ferry Unit 1, Amendment No. 93.

3. Analog trip system

The analog transmitter trip system (ATTS) is a new design for portions of the system instrumentation of the Reactor Protective System (RPS) of Boiling Water Reactors. It was developed by the General Electric Company (GE) and is being supplied as original equipment in later built BWRs (e.g., BWR 6). GE developed the ATTS to offset operating disadvantages of the digital sensor switches of the original safety system instrumentation. The principal objective of the ATTS is to improve sensor intelligence and reliability while enhancing testing procedures.

The design was adapted to Browns Ferry Unit 2 to replace the existing mechanical switches that sense drywell and reactor pressures with analog loops and to modify the reactor water level indication loops to improve the reliability, accuracy and response time of the instrumentation. Change in design basis, protective function, redundancy, trip point, and logic would not be involved or modified as a result of the equipment changes.

Basically, the licensee is proposing to replace Barton, Barksdale, Static-O-Ring, and Yarway instruments with Rosemount analog pressure transmitters and Rosemount analog trip units. Along with the system enhancement offered by the new electronic instrumentation, the licensee proposed to extend the maximum calibration interval to "once an operating cycle." This was based on the high reliability of the analog instrumentation systems.

The various calibration intervals (not the same as functional test intervals) being used at the plant are:

1) Once every 7 days
2) Once every 3 months
3) Once every 6 months
4) Once every 18 months
5) Once each refueling outage

The channel calibration once per operating cycle is less conservative than the present requirement for calibrations of some systems once every 18 months.

It has come to our attention that the duration of an operating cycle may not be adequately defined. Mid-cycle shutdown may occur such that an operating cycle may be extended well beyond the 18-month period which has been previously considered to be the longest operating cycle. The operating cycle time is dependent on the reload fuel design, which can vary between 12 and 18 months.

The primary factor in setting the calibration intervals is the drift of the transmitters and trip units. The total loop accuracy and the total loop drift are added to obtain the trip setpoint. In many cases, the manufacturer's specifications only provide drift values for 6 to 12 month intervals. These drift values must now be extrapolated linearly to provide for 18 months or longer calibration intervals.

Based on the above information, we concluded that the Technical Specification changes extending the calibration frequencies to "once/operating cycle" are acceptable if these calibration frequencies/intervals are limited to 18 months maximum. This limitation of once/operating cycle not to exceed 18 months for calibration intervals applies to the analog pressure transmitters and analog alarm units only and not to the mechanical pressure switches and their associated alarm units.

By letter dated April 29, 1986, TVA submitted supplement 3 to the amendment request dated August 23, 1984, which made the change from once per operating cycle to a minimum frequency of once per 18 months. Based on that supplement and our review we conclude that the proposed modifications are acceptable.

4. Scram permissive pressure switches

This has been covered in Section A above.

5. Drywell temperature and pressure

The drywell temperature and pressure surveillance instrumentation is being upgraded this outage to provide qualified, more reliable instrumentation. The TS, Tables 3.2.F and 4.2.F, have been revised to reflect new instrument numbers for the new upgraded drywell temperature and pressure instrumentation. The surveillance requirements remain the same. We have reviewed the proposed changes and based on our review find them acceptable.

6. TMI Action plan items (NUREG-0737)

In November 1980, the staff issued NUREG-0737, "Clarification of TMI Action Plan Requirements," which included all TMI Action Plan items approved by the Commission for implementation at nuclear power reactors. NUREG-0737 identifies those items for which Technical Specifications are required. A number of items which require Technical Specifications were scheduled for implementation after December 31, 1981. The staff provided guidance on the scope of Technical Specifications for all of these items in Generic Letter 83-36. Generic Letter 83-36 was issued to all Boiling Water Reactor licensees on November 1, 1983. In this Generic Letter, the staff requested licensees to:

a. review their facility's Technical Specifications to determine if they were consistent with the guidance provided in the Generic Letter, and

b. submit an application for a license amendment where deviations or absence of Technical Specifications were found.

By letter dated August 23, 1984, as supplemented, TVA responded to Generic Letter 83-36 by submitting Technical Specification change request for Browns Ferry Unit 2. This evaluation covers the following TMI Action Plan items:

Noble Gas Effluent Monitor (II.F.1.1)

The licensee has supplemented the existing normal range monitors to provide noble gas monitoring in accordance with TMI Action Plan Item II.F.1.1. The proposed Technical Specifications for Noble Gas Effluent Monitor are consistent with the guidelines provided in Generic Letter 83-36. Therefore, we conclude that the TSs for Item II.F.1.1 are acceptable.

Sampling and Analysis of Plant Effluents (II.F.1.2)

The guidance provided by Generic Letter 83-36 requested that an administrative program should be established, implemented and maintained to ensure the capability to collect and analyze or measure representative samples of radioactive iodines and particulates in plant gaseous effluents during and following an accident. The licensee has proposed TSs that are included with the TSs for Surveillance Instrumentation. The proposed TSs for sampling and analysis of plant effluents meet the intent of our guidance. Therefore, the proposed TSs are acceptable.

Drywell High-Range Radiation Monitor (II.F.1.3)

The licensee has installed two drywell radiation monitors in Browns Ferry Unit 2 that are consistent with the guidance of TMI Action Plan Item II.F.1.3. Generic Letter 83-36 provided guidance for limiting conditions for operation and surveillance requirements for these monitors. The licensee proposed TSs that are consistent with the guidance provided in Generic Letter 83-36. Therefore, we conclude that the proposed TSs for Item II.F.1.3 are acceptable.

Drywell Pressure Monitor (II.F.1.4)

Browns Ferry Unit 2 has been provided with two wide range channels for monitoring drywell pressure following an accident. The licensee has proposed TSs that are consistent with the guidelines contained in Generic Letter 83-36. Therefore, we conclude that the proposed TSs for drywell pressure monitors are acceptable.

Suppression Pool Water Level Monitor (II.F.1.5)

The suppression pool water level monitors at Browns Ferry Unit 2 provides the capability required by TMI Action Plan Item II.F.1.5. The proposed TSs contain limiting conditions of operation and surveillance requirements that are consistent with the guidance contained in Generic Letter 83-36. Therefore, we conclude that the proposed TSs for suppression pool water level monitors are acceptable.

7. Testable Penetrations

Modifications are being made to the flange side of 14 containment isolation valves which cannot be isolated from primary containment to be tested. This modification will provide two gaskets with a pressure tap between the gaskets to allow the flange to be leak tested. Operability of the valve will not be affected by this modification. Fourteen new testable penetrations resulted and they were added to the table of testable penetrations with double o-ring seals (Table 3.7.B). New surveillance requirements are also being added. This change was previously approved for Unit 3 by Amendment No. 78 dated August 27, 1984.

Several editorial changes were also made to this table. They include revising the identification name on several penetrations, adding a penetration that was tested but was inadvertently left out of the table and removing penetration X-213A which no longer exists. These changes are purely administrative. Other minor corrections to this table were also made. Penetration X-35G was listed in this table for "T.I.P Drives" and is being revised to reflect that it is a "Spare." The drywell head is being added to this table. It was inadvertently not listed, but was included in the surveillance program. We have reviewed the proposed changes and find that the changes bring Table 3.7.B into conformance with 10 CFR 50 Appendix J for all testable penetrations with double o-ring, and are acceptable.

8. Redundant Air Supply to the Drywell

This proposed change was removed by supplement 2 to the amendment request dated December 30, 1985.

9. Demineralized Water Isolation Valve

The TSs are revised to delete primary containment isolation valve 2-1143 of the demineralized water system. This valve isolated 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. No safety-related functions will be adversely affected by disconnecting this line. This was previously approved for Unit 3 by Amendment No. 78 dated August 27, 1984.

We have reviewed this change and find that the TS change replacing the valve by a cap that will not leak is acceptable.

10. Residual Heat Removal (RHR) Head Spray

Two isolation valves on the residual heat removal head spray line were removed from Unit 2. The head spray line was removed and the penetration capped. The TS are being revised to remove these valves from the table of valves to be tested. The change deletes primary containment isolation valves 74-77 and 74-78 of the RHR system head spray from Tables 3.7.A and 3.7.F. The removal of the head spray line is part of the Intergranular Stress Corrosion Cracking Study being done on Browns Ferry. No safety related functions will be adversely affected by disconnecting this line.

We have reviewed this change and find it acceptable.

D. Administrative Changes

Several administrative changes are being made to the Technical Specifications. These include revising the Table of Contents to reflect the change discussed above, and miscellaneous editorial changes such as to delete obsolete references, change bases to reflect the changes to the Technical Specifications, correct page numbers, correct typographical errors, etc. The surveillance requirements for the personnel air lock is being changed to be consistent with the surveillance for Units 1 and 3. The proposed change includes deletion of the reference to safety valves in conjunction with relief valves. The safety valves with unpiped discharge have been removed and replaced with relief valves.

3.0 ENVIRONMENTAL CONSIDERATIONS

This amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes 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.

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

Attachment:
Evaluation

Principal Contributors: W. Brooks, G. Schwenk, J. Mauk, C. Patel, and M. Grotenhuis

Dated: August 19, 1986

References

1. Browns Ferry Nuclear Plant Reload Licensing Report, Unit 2, Cycle 6; TVA-RLR-002, July, 1984, as supplemented.
2. L. T. Mayhue, "QUAD + Demonstration Assembly Report", WCAP-10507 (Proprietary), March, 1984.
3. GESTAR II, "General Electric Standard Application for Reactor Fuel", NEDO-24011-A-4, January, 1982.
4. B. L. Darnell, et. al, "Methods for the Lattice Physics Analysis of LWR's," TVA-TR78-02A, April, 1978.
5. S. L. Forkner, et. al, "Three Dimensional Core Simulator Methods", TVA-TR78-03A, January, 1979.
6. "Verification of TVA Steady State BWR Physics Methods", TVA-TR79-01A, January, 1979.
7. "BWR Transient Analysis Model Utilizing the RETRAN Program", TVA-TR81-01, December, 1981.
8. Browns Ferry Nuclear Plant Reload Licensing Report, Unit 3, Cycle 6; TVA-RLR-001, January 1984.

ATTACHMENT

EVALUATION RELATING TO TOPICAL REPORT WCAP-10507

QUAD + DEMONSTRATION ASSEMBLY REPORT

1.0 INTRODUCTION

Westinghouse Nuclear Energy Systems has prepared a report, WCAP-10507, "QUAD - Demonstration Assembly Report" and submitted it to the NRC staff for information. Since TVA has referenced this report in its application for the Cycle 6 reload of Browns Ferry Unit 2, the staff has performed a "mini-review" of the report to evaluate the impact of including four of the QUAD + assemblies in the core as Lead Test Assemblies (LTAs). All aspects of the assembly performance are evaluated except that of thermal-hydraulic stability. That aspect is the subject of a separate evaluation. The evaluation follows.

2.0 EVALUATION

The QUAD + assembly has been designed to be a reload bundle for BWR/3 through BWR/6 cores with either "C" or "D" lattice designs. It is intended to provide reduction in fuel cycle costs along with increased thermal margins. Care has been taken to make the QUAD + assembly compatible with currently used BWR bundles, particularly the P8x8R design. Details of the design of the QUAD + assembly are held to be proprietary information by Westinghouse.

The report also includes a set of constraints to be used when inserting QUAD + assemblies into a core as lead test assemblies (LTAs). These include:

1. The QUAD + demonstration assembly will not become a lead assembly during normal operation.

2. The QUAD + demonstration assembly will not become limiting under transient conditions.

3. One QUAD + demonstration assembly should be placed adjacent to a Local Power Range Monitor (LPRM) string.

4. QUAD + demonstration assemblies should be loaded quarter-core symmetric.

5. QUAD + demonstration assemblies will not be loaded less than one row away from the analytically determined potential dropped rod.

6. QUAD + assemblies should preferably not be loaded next to control rods which are inserted in the power range of operation during the first cycle.

2.1 Fuel Mechanical Design

The QUAD + assembly is designed to have the same length as the standard BWR assembly but has slightly larger lateral dimensions. The QUAD + channel design has improved creep resistance compared to the standard design which ensures that an adequate gap between assemblies is maintained throughout core residence time to permit unhampered control rod movement. The upper and lower end fittings of the QUAD + design interface with the core internals in the same manner as those of the standard design.

The QUAD + assembly contains more fuel rods than the standard assembly. Each rod is smaller in diameter than the standard rod and is surrounded by Zircalloy cladding which has been specially treated to improve corrosion resistance. Six-inch blankets of natural uranium are provided at the top and bottom of the fuel stack and gadolinia is used in selected rods to improve radial power distribution and to control assembly reactivity. Top and bottom structures are designed to be compatible with the core internals. Grid spacers have been designed for low flow resistance and improved thermal performance. Fuel rod integrity is assured by evaluation to design criteria which prevent excessive fuel temperatures, excessive internal rod gas pressures due to fission gas release, clad flattening, fatigue, corrosion above clad material removal limits, and excessive cladding stresses and strains during normal operation and anticipated transients. The Westinghouse PAD fuel performance code was used for the analyses. This code has been approved for use with PWR fuel and we find its use for QUAD + fuel acceptable for lead test assemblies. This conclusion is based on the fact that large margins will be maintained between safety limits and expected fuel duty for the LTAs. The design evaluations show that the QUAD + fuel meets all the design criteria with margin.

2.2 Nuclear Design

The nuclear design of the QUAD + assemblies is described in the report. The assemblies were designed to be as nearly the same as the P8x8R replacement fuel as feasible. The assembly design and comparison calculations were performed with the PHOENIX and POLCA codes. These codes have not been formally reviewed by the staff but information has been provided by Westinghouse to show that the PHOENIX assembly code gives results consistent with their standard design methods. The POLCA code is sufficiently similar to the Westinghouse PALADON code to permit the conclusion that the 3-D comparisons are acceptable, particularly since the QUAD + assembly are located in non-limiting positions.

Comparisons were made between the two assemblies for:

° assembly reactivity (K_∞ vs exposure)

- local peaking factor
- void coefficient
- moderator temperature coefficient
- Doppler coefficient
- cold rodded and unrodded reactivity
- rod worth as a function of void content
- delayed neutron fraction and prompt neutron lifetime.

These calculations demonstrated that the QUAD + assembly characteristies were similar of those of the P8x8R assembly it is designed to replace, or were conservative with respect to it. Three dimensional calculations were performed with a QUAD + assembly replacing a standard assembly to confirm that such replacement has no significant effect on core behavior. The QUAD + assembly has a slightly flatter end-of-cycle axial power distribution than the standard assembly due to a smaller void coefficient in the former. LPRM readings near the QUAD + assembly were within 1 to 3 percent of those for a standard assembly assembly - well within the LPRM uncertainty. We conclude that substitution of four QUAD + assemblies for four standard assemblies will have negligible effect on the neutronic behavior of the core.

2.3 Thermal-Hydraulic Analysis

Acceptability of the thermal-hydraulic design is based on hydraulic compatibility of the QUAD + design with the 8x8R standard design and on acceptable CPR performance. It is claimed that flow tests have shown that virtually identical pressure drops exist across the two bundle types at rated core flow and power conditions, but no data are presented. Outer bypass flows and in-channel flows are also the same for the assembly types. Hydraulic compatibility is thereby assured. The CPR performance of the QUAD + assembly is calculated with the AA-74 correlation developed by ASEA-ATOM for an 8x8 fuel assembly. This use is supported by the observation that the improved spacer grid design results in extra CPR margin for the QUAD + assembly. The use of the GEXL safety limit value of 1.07 for the QUAD + assembly (used with the AA-74 correlation) is supported by the fact that the convoluted uncertainties of the parameters used in the CPR evaluation are essentially the same for the two correlations. However, the form of the two correlations is different and the conclusion that a limit of 1.07 applies to both may not be valid. Finally the GEXL correlation will be used for the QUAD + demonstration assemblies when operating in the reactor.

The two correlations have been compared for a number of plant operating conditions and shown to give similar results.

In order to obtain additional margin to CPR limits the guidelines listed in Section 1 above are designed to provide a 10-20 percent margin in power between the QUAD + assemblies and the leading assembly under normal operating core conditions.

2.4 Transient and Accident Analyses

2.4.1 Core-Wide Transients

The consequences of core-wide transients depend upon core-wide neutronics parameters, which are not altered significantly by the presence of the four

QUAD + assemblies. Thus the core response is not altered but the transient response of the assemblies themselves must be considered. For slow transients, such as loss of feedwater heater, the change in CPR for the QUAD + assembly is essentially the same as that for the P8x8R assembly. The rapid transients, such as load rejection without bypass, result in larger MCPR changes for the QUAD + fuel relative to the standard fuel. For a typical such transient the change in CPR of a QUAD + bundle could be as great as 8 percent larger than that for the standard bundle. As indicated in Section 4 above a margin of 10 to 20 percent is provided by following the guidelines given in Section 1. In view of the increased change in CPR during transients and the uncertainties in the applicability of the GEXL correlation to the QUAD + assembly we conclude that the generic margin of 10 to 20 percent is not sufficient. We will therefore require cycle specific calculations to assure that a margin of at least 20 percent is present.

2.4.2 Dropped Rod

The QUAD + assemblies will be placed in the core in positions at least one row away from the rod shown by analysis to have the greatest worth in the startup regime where the consequences of the rod drop accident are significant. The QUAD + assembly will thus not be limiting for this event.

2.4.3 Rod Withdrawal Error

The rod worths at power are smaller for QUAD + assemblies than for standard ones. In addition the QUAD + assemblies will be loaded into non-limiting locations. The intent of the demonstration program is to have the QUAD + assemblies in non-rodded locations at power. For these reasons the presence of the QUAD + assemblies will not affect the rod withdrawal error analysis.

2.4.4 Fuel Misloading Event

The mislocation and misorientation of QUAD + assembly has been analyzed. Since it has been designed to have essentially the same reactivity as the corresponding P8x8R assembly the analysis for the latter assembly is applicable. The flatter enrichment distribution factor of the QUAD + assembly result in smaller changes in LHGR and CPR for misorientation events than with the corresponding P8x8R assembly.

2.4.5 Loss of Coolant Accident (LOCA)

The QUAD + assembly has several features which tend to mitigate the consequences of the loss of coolant event when compared to the equivalent P8x8R assembly. These include improved radiation heat transfer characteristics and a thinner channel which is more easily quenched. The lower plate design tends to delay the voiding of the assembly leading to an extended film boiling period. For the same fuel bundle power, the linear heat generation rate in the fuel is lower. These reactors tend to reduce the peak cladding temperature in a LOCA compared to the equivalent P8x8R assembly. Thus it may be concluded that the

LOCA analysis performed for a core loaded with standard assemblies will be applicable to QUAD + fuel and that MAPLHGR limits obtained for the equivalent P8x8R assembly may be conservatively applied to the QUAD + assembly.

3.0 CONCLUSIONS

Based on the review which is described above we conclude that WCAP-10507 presents sufficient information to support the use of up to four QUAD + bundles as demonstration assemblies in BWR/3 through BWR/6 cores provided that:

1. The guidelines presented in Section 4.1.2 of WCAP-10507 are adhered to, and

2. Cycle specific analyses are performed to show that a margin of at least 20 percent in power exists between the QUAD + assembly and the lead assembly when the core is operating at full power, full flow conditions.

Any more extensive loading of QUAD + assemblies into BWRs will be subject to review in considerably greater depth than is described in this evaluation.