SUPPLEMENT NO. 2

TO THE

SAFETY EVALUATION

BY THE

DIRECTORATE OF LICENSING

U. S. ATOMIC ENERGY COMMISSION

IN THE MATTER OF

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT

UNITS 1, 2, AND 3

DOCKET NOS. 50-259, 260, AND 296

ISSUANCE DATE: June 22, 1973

INTRODUCTION

The Atomic Energy Commission's Safety Evaluation Report (SER) on the Browns Ferry Nuclear Plant Units 1, 2, and 3, dated June 26, 1973, identified certain matters as requiring additional information from the applicant or that were still under review by the Regulatory staff. Supplement No. 1 to the SER, dated December 21, 1972, updated the SER in many areas and also addressed those matters which were cited in the Advisory Committee on Reactor Safeguards (ACRS) "Report on Browns Ferry Nuclear Plant Unit 1," September 21, 1972. This report which is attached as Appendix A to Supplement No. 1 contains the findings of the ACRS upon completion of its review of the Browns Ferry Nuclear Plant Unit 1 at its September 14-16, 1972, meeting.

The purpose of this Supplement No. 2 is to further update the SER and to address a generic review matter which has been under consideration by the Regulatory staff during the past year. Part A of this supplement addresses items that constitute our further updating of the SER and Part B addresses the generic item. Each of the sequentially numbered items in Part A of this supplement contains a specific reference to the section of the SER that is being updated by the material provided in this supplement.

PART A: FURTHER UPDATING OF THE SER

Item 1: Vibration Control (Section 3.5.3)

Supplement 1 to the SER concluded that the preoperational vibration test program for the Browns Ferry Plant needed to qualify Unit 1 as the prototype for reactor internals vibration testing is acceptable subject to receiving the predictive vibration analysis. The applicant submitted the predictive vibration analysis in

Amendment 46 to the FSAR. The staff's review resulted in modifications which have been documented by the applicant and we conclude that the vibration test program is satisfactory.

Item 2: Standby Gas Treatment System (Section 7.2.1(5))

The SER stated that we would require the applicant to modify the design of the system to provide for concurrent starting of the fans. The applicant has now modified the system design and completed those field changes necessary for concurrent and simultaneous start of both fans. We conclude that the system design is acceptable.

Item 3: Control Rod Drop Accident (Section 9.4 of the SER)

In Supplement No. 1 to the Safety Evaluation, we indicated that design modifications are required to limit the consequences of a control rod drop accident to a peak fuel enthalpy less than 280 calories per gram and to limit the radiation dose to less than the guideline values of 10 CFR Part 100. Amendments 48 and 50 of the FSAR contain a description of the applicant's proposed Rod Sequence Control System (RSCS) and of the modifications which were required to limit the peak fuel enthalpy which could result from revised design basis rod drop accident. Between the beginning of life (BOL) and 6500 MWD/T, the RSCS will control within sequence A, the selection of rods for withdrawal from four separate rod groups. These will be designated as A_{12}, A_{34}, B, and C. The group C rods consist of 32 selected control rods which circumscribe the interface between the Type 2 and Type 3 gadolinia bearing fuel in the reactor. Final evaluation by the applicant with regard to the control rod drop accident established that the group C rods had reactivity worths for the 50 percent control rod density pattern at BOL (all group A_{12} and A_{34} rods fully withdrawn) greater than 0.0145 $\Delta\kappa$ which would result in peak fuel enthalpies exceeding the 280 cal/gm design limit. Therefore, additional restrictions must be placed on group C control rods. The applicant has estimated that at 10 percent power, these restrictions may be removed since the maximum worth of any group C rod is less than 0.01 $\Delta\kappa$. The system design will prohibit the group C rods from being withdrawn by interlocks and these rods will remain in the fully inserted position until the reactor power reaches 30 percent of full power. After achieving an exposure of 6500 MWD/T, the high cross section gadolinium isotope has been depleted and the radial variation in gadolinia which required control of group C rods beyond 50 percent rod density is no longer required.

Until the core exposure exceeds 6500 MWD/T, only one withdrawal sequence is possible. After the core exposure exceeds 6500 MWD/T the group C control rods will be rewired back into group B, and thereafter either sequence A or B can be used during reactor startup operations. For fuel cycles after the first cycle, the number of Group "C" rods to be controlled by the RSCS and the exposure limit are not known. Therefore, further staff review effort will be needed for subsequent fuel cycles.

We have reviewed the criteria and design of the rod sequence control system. Certain design modifications and installation criteria have been required to assure that the RSCS design will be acceptable. In addition, the applicant has stated that an annunciator subsystem will be added to the RSCS prior to operation of Unit 1. This subsystem will provide an alarm signal to the operator when the RSCS has been bypassed due to some malfunction or fault in the sequence logic system. We find this to be acceptable.

Our consultant, the Brookhaven National Laboratory, has performed calculations which tend to confirm the General Electric Company calculations of the peak fuel enthalpy resulting from the revised design basis control rod drop accident. These revised methods and models are given in the General Electric Company Topical Report "Rod Drop Accident Analysis For Large BWR's," NEDO-10527 and in its Supplements Nos. 1 and 2. Our generic review of these reports is continuing; however, we have completed our review of the General Electric Company transient techniques and conclude they are acceptable.

The applicant has indicated in Amendment 48 that the postulated control rod drop accident with a single operator error and with the RSCS in operation, will result in a peak fuel enthalpy of 230 calories per gram for core exposures less than 6500 MWD/T and 215 calories per gram for core exposures greater than 6500 MWD/T.

The applicant has stated that the postulated control rod drop accident will result in a perforation of 600 fuel rods. In Section 9.4 of the Safety Evaluation, we evaluated the consequences of the control rod drop accident based on 330 fuel rod perforations. The doses calculated for this postulated accident were 3.6 rem thyroid and less than one rem whole body at the exclusion distance and 59 rem thyroid and less than 1 rem whole body at the low population zone distance. The above cited doses are directly proportional to the number of fuel rods which perforate. Therefore, the 600 fuel rod perforations will result in an 80% increase in the above cited doses for the exclusion distance and low population zone distance and are well below the guidelines of 10 CFR Part 100.

We conclude that the RSCS will provide adequate protection to limit the consequences of a control rod drop accident, with a single operator error, to doses that are well within the guidelines of 10 CFR Part 100 and to limit the peak fuel enthalpy to less than 280 calories per gram. The Browns Ferry Nuclear Plant Unit 1 Technical Specifications will require that the Rod Sequence Control System be operable during all reactor operations below 30% power level. The control rod Technical Specification scram time to 90% insertion will be 5 seconds which has been reflected in the revised rod drop accident analyses.

PART B: ADDITIONAL REVIEW MATTERS

Item 1: Fuel Densification

During the early stage of reactor operation the uranium oxide pellets in the fuel pins undergo an increase in density due to micro-structural changes in the uranium oxide at high temperatures and the disappearance of small voids from the uranium oxide matrix during irradiation. A description of the fuel densification phenomena and its effects on fuel performance is given in the "Technical Report on Densification of Light Water Reactor Fuels," issued by the Regulatory staff, U. S. Atomic Energy Commission on November 14, 1972. The applicant has indicated that the fuel specifications, design, and fabrication are as described in the General Electric Company topical report, "Densification Considerations in BWR Fuel Design and Performance," NEDM-10735 dated December 1972. This topical report is currently under review by the Regulatory staff. Additional information has been requested by the staff to permit an independent evaluation of the consequences of the fuel densification on Browns Ferry type boiling water reactors. The applicant has submitted additional information in Supplements 1, 2, and 3 dated 4/5/73, 5/9/73, and 6/19/73 respectively. The significant areas of review include the effects of densification on gap conductance, clad creepdown, and the power spike due to axial gaps. The staff expects to complete its review of the fuel densification matter for the Browns Ferry Nuclear Plant in July 1973.

The applicant presented a motion to the Atomic Safety and Licensing Board (ASLB) on May 25, 1973, requesting an order authorizing the Director of Regulation to make appropriate findings and issue a license for operation up to 75% of rated power. The ASLB issued the order on June 15, 1973.

We have concluded, as a result of our fuel densification review to date including independent staff calculations, that operation of the facility at power levels up to 75% of rated power provides more than adequate margin for potential adverse effects of fuel densification.