SUPPLEMENT NO. 1

TO THE

SAFETY EVALUATION

BY THE

DIRECTORATE OF LICENSING

U. S. ATOMIC ENERGY COMMISSION

IN THE MATTER OF

DUKE POWER COMPANY

OCONEE NUCLEAR STATION UNITS 2 AND 3

DOCKET NOS. 50-270, -287

AUG 2 1973

TABLE OF CONTENTS

1.0 INTRODUCTION

1.1 General

Duke Power Company (the applicant) applied for an operating license for the Oconee Units 2 and 3 reactors by application dated June 2, 1969. The Atomic Energy Commission's Regulatory Staff (the staff) subsequently completed its review of the application and issued a Safety Evaluation Report on July 6, 1973. A notice of intent to issue an operating license was published in the Federal Register on August 10, 1972, by the Atomic Energy Commission. No hearing was requested.

On November 14, 1972, the Regulatory Staff issued a report entitled, "Technical Report Densification of Light Water Reactors Fuels"(1)* which resulted from the staff's consideration of the Ginna fuel densification phenomenon. Based upon the findings in this report the staff requested on November 20, 1972 that the applicant provide analyses and relevant bases, in accordance with the densification report,(1) that determine the effects of fuel densification on normal operation, transients and accidents for the three Oconee Units. On January 16, 1973 the applicant filed a response to the request(2,3) for Oconee Unit 1 as a lead plant for this evaluation. On March 14, 1973, the staff requested additional information. The applicant filed a response to this request on April 13, 1973.(4,5) On June 29 the applicant filed a response to these requests specifically for Oconee Unit 2.(11)

*Numbers in () refer to references listed in Section 6.0.

The staff's technical review of fuel densification as it applies to Oconee Unit 2, and the technical evaluation of the applicant's safety analysis of steady state operation, operating transients and postulated accidents taking into account the effects of densification are presented in this supplement.

This evaluation relies upon the July 6, 1973 Regulatory staff report "Technical Report on Densification Report of Babcock & Wilcox Reactor Fuels"[6] which concluded that B&W's fuel densification models are in compliance with the staff's initial densification report[1].

The staff has concluded that the operation of Oconee Unit 2 for the first cycle at power levels up to 100 percent of full power, in accordance with the Technical Specifications, will not present an undue risk to the health and safety of the public. Oconee Unit 3 has not been evaluated since as built data is not available at this time. However, the staff will evaluate Oconee Unit 3 prior to operation.

1.2 Scope of Review

The essential elements that must be considered in evaluating the effects of fuel densification have been set forth in the staff's initial densification report.[1] Since the performance of the facility in steady state operation and during various postulated transients and accidents had been established previously as reported in the Final Safety Analyses Report (FSAR) without the assumption of fuel densification, it was only necessary to evaluate those changes in the analyses and in the results that are attributed to fuel densification. The effects of fuel densification on the steady state operation and on the course of postulated plant transients and accidents were evaluated by the applicant and reviewed by the staff.

The staff reviewed the effects of fuel densification for Oconee Unit 2 using the staff's guidelines, the technical evaluation of the applicant's safety analysis of steady state operation, operating transients and postulated accidents and the generic evaluation[6] of B&W methods for assessing fuel densification and its effects. In the evaluation the applicant appropriately considered the staff guidelines including the effects of instantaneous and anisotropic densification (initial density minus 2σ , and final density 96.5% TD), the assumption of no clad creepdown as a function of core life, and the assumption of an axial gap leading to a power spike. The staff reviewed the effects of fuel manufacturing and reactor operating parameters on the fuel densification mechanism. The generic evaluation of these items is included in Reference 6. The staff reviewed B&W's assumptions, methods, and computer codes used in evaluating the fuel densification effects. The generic evaluation of B&W's models is also included in Reference 6. The mechanical integrity of the fuel cladding and the thermal performance of the fuel were considered in the analyses of steady state operation, operating transients, and postulated accidents as discussed in the following sections.

2.0 MECHANICAL INTEGRITY OF CLADDING

Clad creepdown during the core life is not considered by the applicant in the calculation of gap conductance. This is a conservative assumption since the reduced gap size due to clad creepdown would result in a higher gap conductance and thus in a lower stored energy in the fuel. The staff reviewed the B&W method for calculating the clad collapse time, which is the time required for an unsupported cladding tube to flatten into the axial gap volume caused by fuel densification. On the basis of independent staff calculations and from experience of fuel performance in other reactors, the staff concurred with the applicant that clad collapse is not expected for the Oconee Unit 2 fuel during the first cycle of 11,040 effective full power hours (EFPH). However, the staff concluded that the evaluation model for collapse time calculations contains several deficiencies in its application to Oconee Unit 2. The staff informed the applicant[7] that an acceptable model for collapse time calculations is necessary for subsequent fuel cycles of Oconee Unit 2.

3.0 EFFECTS OF DENSIFICATION ON STEADY STATE AND TRANSIENT OPERATION

3.1 General

Fuel densification can affect the steady state operation because of axial gaps in the fuel column that results in local neutron flux spikes and an overall increased linear heat rate. An additional effect occurs in the transient analyses since, due to a lower gap conductance, the fuel has a higher initial stored energy and a slower heat release rate during the transient.

On the basis of evaluations of the effects of fuel densification the Oconee Unit 2 reactor will be operated with more restrictive limits on control rod patterns and position than originally proposed, and with a reduced maximum linear heat generation rate. The changes consider the effects of local peaking caused by gaps in the fuel pellet stack and changes in the gross peaking factors, primarily axial, which can be achieved by more restrictive operation of control rods.

The effects of densification on power density distributions have been calculated using models in conformance with those discussed in Section 4 of the staff densification report.(1) The primary calculations used the models and numerical data of the Westinghouse power spike model as described in Appendix E of that report, except that the initial nominal density used was [ ] (the minimun density of the three batches), and the probability of gap size was changed to conform to that recommended by the staff.(1)

---

*[ ] Brackets denote data known by the staff and considered proprietary to the applicant and specified in references 4 and 5 to this report.

The calculations by the applicant take into account the peaking due to a given gap, the probability distribution of the peaks due to the distribution of gaps, and the convolution of the peaking probability with the design radial power distribution. The calculations result in a power spike factor that varies almost linearly with core height and reaches a maximum value of 1.15 at the top of the core. The overall calculation falls within the range examined[12,13] by our consultant, Brookhaven National Laboratory, in conjunction with reviews of other models.

A normalized shape for the power spike factor is derived from power spikes caused by different gap sizes at various axial locations. The normalized shape is then used in conjunction with various axial power shapes to determine the axial location at which the decrease in DNBR due to the superimposed power spike is maximized. These calculations also include the increase in average linear heat generation rate from 5.656 Kw/ft to 5.774 Kw/ft due to the reduced fuel column height based on the instantaneous densification from the minimum initial density of [ ] theoretical density (TD) to a final density of 0.965 TD.[1] The reactor operating limits, which are part of the Technical Specifications for Oconee Unit 2, are based on maximum linear heat generation rate through the reactor power vs axial offset correlation.

3.2 Fuel Rod Thermal Analysis

The applicant uses the B&W computer code, TAFY[10], to calculate gap conductance, fuel temperature, and stored energy for the Oconee Unit 2 fuel, which in turn are used in the safety analyses. To demonstrate the applicability of the TAFY code for the evaluation of the Oconee Unit 2 fuel thermal behavior, the applicant compared TAFY predicted fuel temperatures and gap conductance with experimental data.

The staff reviewed the TAFY code and concluded that realistic and/or conservative assumptions have been used for modeling of the physical phenomena incorporated into the code (thermal expansion, fuel swelling, sorbed gas release, fission gas release), with two exceptions: (1) partial contact between the clad and fuel and (2) formation of a central void due to fuel restructuring on the basis of columnar grain growth at a temperature of 3200°F. Details of the staff's evaluation of the TAFY code and its application to Oconee Unit 2 type fuel rods are given in Reference 6.

Because of the two exceptions noted above, the staff required the applicant to analyse the fuel thermal performance using a 25% reduction in gap conductance and taking no credit for fuel restructuring. This analysis[9] resulted in a reduction in the peak linear heat rate at which centerline fuel melting would occur from 22.2 Kw/ft before densification to 19.8 Kw/ft after densification was conservatively taken into account. The reactor protection system prevents fuel centerline melting from occurring for all anticipated transients. This is accomplished by proper setting of the reactor trip as a function of power level and axial power imbalance. These settings will be given in the Technical Specifications.

3.3 Steady State and Loss-of-Flow Transient

The effect of fuel densification on the departure from nucleate boiling ratio (DNBR) during steady state operation was analyzed by both the applicant and the staff. The staff's independent calculations are described in Reference 6. The results show that the steady state minimum DNBR decreases due to an increase in the surface heat flux resulting from fuel densification. To assess the amount of reduction in DNBR margin, the applicant reanalyzed the steady state operating and design overpower conditions with an assumed axial power shape that peaked near the core outlet rather than with the symmetrical reference design power shape described in the FSAR. The outlet shape, though not achievable in operation, produces the largest possible DNBR penalty from fuel densification, because the point of minimum DNBR is shifted toward the top of the hot fuel rod where the densification induced power spike is the largest. The application of this large power spike at the point of minimum DNBR produces the greatest degradation in DNBR. Using this outlet axial power peak the applicant computed a 6.1% reduction in DNBR from the 1.55 value reported in the FSAR without the effects of densification. The applicant has proposed more stringent control rod positions and offset limits to compensate for the loss in DNBR margin. This is acceptable to the staff.

B&W also reanalyzed the loss of flow transient that would result from a loss of electrical power to the reactor coolant pumps taking into account the effects of fuel densification. The results show that the minimum DNBR during the transient decreased, due to local flux increases caused by fuel densification. The previously calculated minimum DNBR during the transient was 1.60 whereas with the densification the minimum DNBR is calculated to be about 1.53.

The densification effects that could aggravate the consequences of the loss-of-flow transient are the increase in the steady state fuel temperature (stored energy), increase in heat flux, and a decrease in gap conductance. The increase in fuel temperature provides more stored heat in the fuel which must be removed during the transient; the higher heat flux provides greater initial enthalpy in the coolant channel. The decrease in gap conductance delays the removal of heat from the fuel resulting in a higher ratio of heat flux to channel flow during the transient and thus a lower DNBR.

3.4 Other Transients

The following other transients have been reviewed to determine whether the effects of densification have resulted in significant changes in their consequences:

Control Rod Withdrawal Incident

Moderator Dilution Incident

Control Rod Drop Incident

Startup of an Inactive Reactor Coolant Loop

Loss of Electrical Power

In the applicant's FSAR these transients were calculated to result in a DNBR in excess of 1.3, or their consequences were shown to be limited to acceptable values by limits to be set forth in the Technical Specifications. The staff has reviewed these transients taking into account the effects of fuel densification and agrees with the applicant that they would not result in a reduction of the core thermal margin, i.e., a DNBR less than 1.3.

3.5 Conclusions

The effects of fuel densification on steady state and transient operation have been evaluated by the applicant and reviewed by the staff.

The effect on steady state operation, mostly due to local increases in thermal neutron flux and heat generation, is to require more restrictive limits on control rod positions and offset limits in the Technical Specifications for Oconee Unit 2. In order to prevent fuel melting the maximum allowable linear heat generation rate has been reduced from 22.2 Kw/ft to 19.8 Kw/ft. The overpower trip limit has been reduced from 114 percent to 112 percent such that a DNBR greater than 1.3 is maintained for steady state and during transient conditions.

The staff concluded on the basis of its review that the potential effects of fuel densification on the steady state and postulated transient operation have been evaluated in an appropriate manner and are acceptable for the period of operation proposed.

4.0 ACCIDENT ANALYSES

4.1 General

Analyses of the consequences of various postulated acci'ents were presented in the FSAR for the Oconee Unit 2. The accidents evaluated were:

(1) Locked Rotor

(2) Loss-of-Coolant (LOCA)

(3) Control Rod Ejection

(4) Steam Line Rupture

(5) Steam Generator Tube Rupture

(6) Fuel Handling

(7) Waste Gas Tank Rupture

Since fuel densification will affect the consequences of the first four postulated accidents they have been reanalyzed by the applicant and reevaluated by the staff. Results of the first three accidents (locked rotor, loss-of-coolant, and control rod ejection) are presented in separate parts of this section. The steam generator tube rupture, waste gas tank rupture, fuel handling and steam line rupture accidents are discussed below.

Changes in the fuel pellet geometry can cause the stored energy in the fuel pellet to increase by the mechanisms discussed in Section 3.0 of this report. Potential increases in local power due to the formation of axial gaps are discussed in Section 3.1. Both of these effects are accounted for in the evaluation of accidents.

The radiological consequences of accidents were independently calculated by the staff. The results of the staff's calculation of the radiological consequences of accidents were presented in the Oconee Unit 2 Safety Evaluation report dated July 6, 1973. The radiological consequences would not increase as a result of fuel densification, although the transient performance of the fuel rods can change as a result of fuel densification. It is the latter factor that is discussed in the following sections.

The staff evaluation of the radiological consequences of a waste gas decay tank failure was based on an assumed quantity of gas in the tank limited by the Technical Specification. For the steam generator tube rupture accident, the assumed quantity of reactor coolant activity is consistent with the Technical Specification limits on maximum permitted reactor coolant system acitivity. Fuel densification will not affect the consequences of these accidents.

The postulated refueling accident assumes the dropping of a fuel assembly in the spent fuel pool or transfer canal. The fuel rods are assumed to be approximately ambient temperature during the postulated accident. Therefore, the direct effects of fuel densification will not affect the consequences of this postulated accident. The potential for mechanical failure of a flattened rod might be different from that of a normal rod; however, since the staff evaluation has been based on the conclusion that no clad collapse will occur during the fuel cycle (Section 2.0), this potential change in fuel rod characteristics was not considered. Furthermore, all of the rods in the dropped assembly are assumed to fail.

The steam line break accident was analyzed by the applicant in the FSAR without the effects of fuel densification. That analysis showed that the worst consequences from this accident would result at the end of life (EOL) of the core. Since the DNBR margin is higher at the EOL, including the effects of fuel densification, the staff does not expect that the thermal limits will be more severe than those presented in the FSAR.

4.2 Locked Rotor Accident

The reactor coolant system for Oconee Unit 2 consists of two loops; each return from the steam-generator to the reactor consists of two cold legs, i.e., a total of four reactor coolant pumps are used. Locked rotor accidents are characteristically less severe for 4 pump plants than for 3 or 2 pump plants.

The analysis of the locked rotor accident was originally presented in Section 14 of the FSAR. The transient behavior was analyzed by postulating an instantaneous seizure of one reactor pump rotor. The reactor flow would decrease rapidly and a reactor trip would occur as a result of a high power-to-flow signal. The core flow would reduce to about three fourths its normal full-flow value within two seconds.

The temperature of th reactor coolant wc ld increase, causing fluid expansion with a resultant pressure transient which would reach a peak of approximately 15 p: above nominal. e applicant computed a maximum cladding temperature of 1380°F at about 4.5 seconds for this accident.

The staff performed independent calculations for this postulated accident using Oconee Unit 1 parameters. The results of these calculations are discussed in Reference 6.

4.3 LOCA Analysis

The B&W evaluation model described in the AEC Intermin Acceptance Criteria and Amendments for Emergency Core Cooling Systems was used by the applicant to evaluate the loss-of-coolant accident (LOCA) for Oconee Unit 2. The analysis was performed with the B&W code CRAFT for the blowdown period and the THETA code for the fuel rod heat up. The applicant's LOCA analysis without the assumption of fuel densification is reported in the Oconee FSAR based on the 8.55 ft^2 split break in the cold leg at the pump discharge as the limiting break size and location.(8)

During the blowdown period the gap conductance, reduced due to fuel densification according to the staff requirements, could cause the core average fuel pellet temperature to increase, but CRAFT calculations show that the temperature experiences only a very small change.

Since in the initial analysis an average core temperature was used that is higher than the average core temperature resulting from the decreased gap conductance, the applicant concludes that the limiting break size and locations do not change due to fuel densification.

The effects of fuel densification on the reflood calculations is small, since the gap conductance is much larger than the film coefficient (cladding surface to coolant) during reflood. The film coefficient is thus limiting with regard to heat transfer and cladding temperature.

The applicant performed the LOCA analysis with an axial power shape that peaks [ ] below the core midplane and a corresponding axial peaking factor of $F^Z = 1.816$ which includes an axial uncertainty factor of 1.024 and a local factor of 1.026 accounting for the effect of the grid structure on the axial peak. This particular flux shape results in the highest linear heat rate and occurs during the control rod maneuvering resulting from the 4-day design basis transient. The design basis transient is defined as a 100% -30% -100% transient, consisting of operation at 100% power, reduction to 30% power, operation at 30% power for about 8 hours, and return to 100% power.

The THETA calculations were performed with the staff requirements for initial fuel pellet density assumptions. However, instead of imposing a power spike due to a fuel column gap at the peak axial power [ ] below core midplane the applicant used an equivalent radial multiplier over the entire length of the fuel pin which leads to a higher calculated peak cladding temperature of approximately 10°F. A hot channel factor of F_{HC} = 1.014 was used in the calcuations. The radial peaking factor, F^R, including an uncertainty factor of 1.05 was varied until the calculated maximum cladding temperature approached the 2300°F limit. Using the gap conductance as calculated with the TAFY code described in Section 3.2 a clad temperature of 2283°F was reached with a maximum linear heat rate of 18.2 Kw/ft, which, therefore, is the maximum allowable linear heat generation rate for the Oconee Unit 2 reactor.(9) In order to accommodate a possible quandrant tilt of 5% during this design basis transient the allowable heat rate is further reduced to 16.38 Kw/ft. The maximum allowable linear heat rate will be controlled by a control rod operating band.

4.4 Rod Ejection Accident

The control rod ejection transient has been reanalyzed(4,5) by the applicant to account for changes in the fuel due to densification. The significant effects of fuel densification are an increase in the initial maximum fuel temperature and a slight increase in average heat flux due to shrinkage of the pellet stack length. In addition, spikes in the neutron power can occur to gaps in the fuel. Calculations have verified that no changes in the basic kinetic response of the core occur due to the small changes in fuel geometry and heat transfer characteristics.

The results of the rod ejection accident at BOL and at EOL without consideration of densification effects have been previously presented in the Oconee FSAR. The staff consultants at Brookhaven National Laboratory (BNL) have performed independent check calculations using appropriate input data and their own computer codes and have confirmed that the results of a rod ejection transient are less severe at EOL than at BOL. Therefore, all calculations by the applicant considering densification effects were done for BOL conditions.

For the full power transient, the control rod reactivity worths available for the assumed ejected rod would be expected to decrease because of the more restrictive insertion limits on the control bank. However, this was not included in the reevaluation, thereby adding additional conservatism to the calculations. The maximum Technical Specification rod worth of 0.65% delta k/k was used for the BOL calculations at full power.

The staff review of the initial fuel temperature for the BOL full power case indicated that a reasonable temperature was used for the assumed conditions, consistent with that used in the LOCA analysis. The neutron power spike effect was included in the reanalysis.

The reexamination of the rod ejection transient considering the effects of densification has resulted in a peak pellet average enthalpy well below the staff's criterion of 280 cal/gm. The maximum center-line fuel temperature reached is well below the assumed melting point of 5080°F, and the maximum clad temperature during the transient is 1510°F. The total number of fuel pins calculated to be in DNB is 28%. The staff review of the rod ejection analysis indicates that reasonably conservative consideration has been given to the effects of fuel densification and that the results are acceptable for this accident.