SUPPLEMENT NO. 3

TO THE

SAFETY EVALUATION

BY THE

DIRECTORATE OF LICENSING

U. S. ATOMIC ENERGY COMMISSION

IN THE MATTER OF

DUKE POWER COMPANY

OCONEE NUCLEAR STATION UNIT 1

DOCKET NO. 50-269

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1.0 INTRODUCTION

1.1 General

Duke Power Company (the applicant) applied for an operating license for the Oconee Unit 1 reactor 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 December 29, 1970. A notice of intent to issue an operating license was published in the Federal Register on January 8, 1971, by the Atomic Energy Commission. No hearing was requested.

On February 6, 1973, after having made appropriate findings, the Commission issued Facility Operating License No. DPR-38 to the Duke Power Company for the Oconee Unit 1. The license is a full power license (2568 MWt).

On November 14, 1972, the Regulatory Staff issued a report entitled, "Technical Report on Densification of Light Water Reactor 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 Oconee Unit 1 facility. On January 16, 1973 the applicant filed a response to the request.(2,3) On March 14, 1973, the staff requested additional information. The applicant filed a response to this request on April 13, 1973.(4,5)

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

The staff's technical review of fuel densification as it applies to Oconee Unit 1, and the technical evaluation of the applicant's safety analyses of steady state operation, operating transients and postulated accidents taking into account the effects of densification are presented in this supplement. Comparison of the Babcock & Wilcox (B&W) calculations (B&W is the applicant's fuel vendor) with the staff's independent analyses is in Appendix A of this report. Appendix A is the staff's generic evaluation of B&W's fuel densification methods and procedures.

The staff has concluded that the operation of Oconee Unit 1 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.

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 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 1 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 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 Appendix A of this report. 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 the Appendix A of this report. 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. The applicant initially proposed to operate the reactor with a combination of prepressurized and unpressurized fuel rods. Because of the increased probability of clad collapse for the latter type all fuel rods subsequently were prepressurized with helium to a pressure of [ ].* On the basis of independent staff calculations and from experience of fuel performance in other reactors, the staff concurs with the applicant that clad collapse is not expected for the Oconee Unit 1 fuel during the first cycle of 7500 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 1. The staff informed the applicant[6] that an acceptable model for collapse time calculations is necessary for subsequent fuel cycles of Oconee Unit 1.

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

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 1 reactor will be operated with more restrictive limits on control rod patterns and motion than originally proposed, and with a reduced maximum linear heat generation rate. The changes consider the effect 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 minimum density of the three batches), and the probability of gap size was changed to conform to that recommended by the staff.(1)

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 1.13 at the top of the core. The overall calculation falls within the range examined 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 linear heat generation rate from 5.66 Kw/ft to 5.74 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 1, 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(9), to calculate gap conductance, fuel temperature, and stored energy for the Oconee Unit 1 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 1 fuel thermal behavior, the applicant compared TAFY predicted fuel temperatures and gap conductances with experimental data.

The staff reviewed the TAFY code and concludes that realistic and/or conservative assumptions have been used for the 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 1 type fuel rods are given in the generic evaluation of B&W's methods for assessing fuel densification and its effects (see Appendix A).

Because of the two exceptions noted above, the staff required the applicant to reanalyse the fuel thermal performance using a 25% reduction in gap conductance and taking no credit for fuel restructuring. This reanalysis[8] 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 20.1 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 are 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 Appendix A. 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 4.46% reduction in DNBR from the 1.55 value reported in the FSAR without the effects of densification. To maintain the same safety margin that existed without the densification considerations as described in the FSAR (i.e., a DNBR of 1.55 at maximum overpower), the applicant proposes to lower the overpower limit from 114% to 112%. 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 primary 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 1.56.

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 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 in the Technical Specifications for Oconee Unit 1. In order to prevent fuel melting the maximum allowable linear heat generation rate has been reduced from 22.2 Kw/ft to 20.1 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 accidents were presented in the FSAR for the Oconee Unit 1. 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 1 Safety

Evaluation report dated December 29, 1970. 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. The assumed quantity is consistent with the Technical Specification limits on maximum permitted reactor coolant system activity. Fuel densification will not affect the consequences of this accident.

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 at 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 1 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 the reactor coolant would increase, causing fluid expansion with a resultant pressure transient which would reach a peak of approximately 15 psi above nominal. The applicant computed a maximum cladding temperature of 1300°F at about 4.4 seconds for this accident.

The staff performed independent calculations for this postulated accident. The results of these calculations are discussed in Appendix A.

4.3 LOCA Analysis

The B&W evaluation model described in the AEC Interim 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 1. 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.(7)

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. Reduced gap conductance during reflood would be a benefit in that the rate of decay heat transferred across the gap to the cladding would be reduced. However, the benefit is not significant since the gap conductance is much larger than the film coefficient during reflood and hence is 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 $F^Z = 1.786$, as discussed in Section 3.1, 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 calculations. 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 2291°F was reached with a maximum linear heat rate of 19.8 Kw/ft; incorporation of the staff recommendations for the TAFY code as described in Section 3.2 reduced the linear heat rate to 18.65 Kw/ft at 2291°F, which, therefore, is the maximum allowable linear heat generation rate for the Oconee Unit 1 reactor.[8] In order to accommodate a possible quadrant tilt of 5% during this design basis transient the allowable heat rate is further reduced by a factor of 1.11 to 16.80 Kw/ft. This 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 due 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 0.50% delta k/k was used for the BOL calculations.

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 of 135 cal/gm, well below the staff's criterion of 280 cal/gm. The maximum centerline fuel temperature reached is 4480°F, well below the assumed melting point of 5080°F, and the maximum clad temperature during the transient is 1305°F. The total number of fuel pins calculated to be in DNB is 13%. 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.

5.0 SUMMARY AND CONCLUSIONS

The effects of fuel densification have been considered in analyses of normal operation, operation during transient conditions, and postulated accident conditions. On the basis of the staff review of the applicant's calculations, and independent calculations performed by the staff and its consultants, the staff concluded that for the period of operation proposed, namely the first fuel cycle:

(1) The effects of densification during steady state and transient operation of the Oconee Unit 1 reactor will not cause the limits on DNBR, cladding strain, and centerline temperatures, to become less conservative than values previously established in the FSAR.

(2) The effects of densification were included in the calculation of fuel rod behavior during postulated loss-of-coolant accidents. The LOCA analysis is acceptable and complies with the June 1971 Interim Acceptance Criteria.

(3) The applicant's omission of the creep down effect, which tends to increase gap conductance with life time, is acceptable.

(4) The Technical Specifications limit the fuel residence time to 7500 effective full power hours of power operation to assure no cladding collapse.

(5) The applicant has adopted the staff recommendations for calculating gap conductances and fuel temperatures (Section 3.2) as they are used in steady state, transient and accident conditions.

(6) Operating restrictions as necessary to assure compliance with items (1) through (4) above have been incorporated into the Technical Specifications.

On the basis of the above summary, the staff concludes that the applicant is in compliance with the staff densification report[(1)] and that Oconee Unit 1 reactor can be operated at power levels up to 100% of rated power with no undue risk to the health and safety of the public.

6.0 REFERENCES

1. "Technical Report on Fuel Densification of Light Water Reactor Fuels," Regulatory Staff, U. S. Atomic Energy Commission, November 14, 1972.
2. "Fuel Densification Report," BAW 10054 Topical Report (Proprietary), January 1973 (Nonproprietary Information in BAW 10055).
3. "Oconee 1 Fuel Densification Report," BAW 1387 (Proprietary), January 1973 (Nonproprietary Information in BAW 1388).
4. "Fuel Densification Report," BAW 10054 - Rev. 2 Topical Report (Proprietary), May 1973.
5. "Oconee 1 Fuel Densification Report," BAW 1387 - Rev. 1 (Proprietary), April 1973.
6. Letter from R. C. DeYoung to R. Edwards, Babcock & Wilcox, dated April 23, 1973, with copy to Duke Power Company.
7. 'Multinode Analysis of B&W's 2568-MWT Nuclear Plants During a Loss-of-Coolant Accident," BAW 10034, Rev. 1, May 1972.
8. Letter from Duke Power Company to A. Giambusso, dated May 14, 1973.
9. "TAFY - Fuel Pin Temperature and Gas Pressure Analysis," BAW 10044, Topical Report, April 1972.

APPENDIX A

TECHNICAL REPORT ON
DENSIFICATION OF BABCOCK & WILCOX REACTOR FUELS

Date: July 6, 1973

REGULATORY STAFF

U. S. ATOMIC ENERGY COMMISSION

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1.0 INTRODUCTION

1.1 General

On November 14, 1972, the Regulatory Staff issued a report entitled "Technical Report on Densification of Light Water Reactor Fuels"[(1)*] which resulted from the staff's consideration of the Ginna fuel densification phenomena. Based upon the findings in this report the staff requested on November 20, 1972 that applicants for licenses of light water reactors provide analyses and relevant bases, in accordance with the densification report, that determine the effects of fuel densification on normal operation, transients and accidents. On January 16, 1973, Babcock & Wilcox (B&W) filed a proprietary generic report,[(2)] BAW-10054, "Fuel Densification Report" which was applicable to all B&W type reactors beginning with Oconee Units 1, 2 and 3 and including Three Mile Island Units 1 and 2; Arkansas Nuclear One Unit 1; Rancho Seco and Crystal River Unit 3. In addition, B&W has submitted through applicants, facility reports describing the effects of fuel densification using calcalutional methods and procedures described in their generic report.[(2)] The first of these facility reports[(3)] was submitted in connection with Oconee Unit 1, the prototype reactor.

The staff has performed a technical review and evaluation of the B&W generic report.[(2)] The results of that review and evaluation are

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

presented in this report and are applicable to all of the above named reactor facilities. Comparisons of B&W's methods and procedures are made with the staff's independent analyses.

To compare specific B&W computer codes to staff computer codes, the specific first cycle fuel characteristics of Duke Power Company's Oconee 1 were used. First cycle fuel characteristics from other B&W designed reactors are similar to those for Oconee 1 and are addressed in the individual reports by B&W and the staff.

The staff has concluded that B&W's methods and procedures for assessing fuel densification and its effects are acceptable provided certain modifications described in detail in the following sections of this report, are employed.

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 densification report.[(1)] Since the performance of a reactor in steady state operation and during various postulated transients and accidents had been established previously and reported in individual facility Final Safety Analysis Reports (FSARs) without the assumption of fuel densification, it is only necessary to evaluate those changes in the analyses and in the results that are attributed to fuel densification. B&W's models for assessing the effects of fuel densification on the steady state operation and on the course of postulated plant transients and accidents have been evaluated and reviewed by the staff on a generic basis.

B&W stated[4,5] that their analysis of the effects of fuel densification is limited to the first fuel cycle, less than 12,000 effective fuel power hours (EEPH) and that collapse of the cladding is not predicted for this period and therefore, the consequences of collapse have not been considered in the evaluation. The staff has concluded independently that the prediction of no collapse is appropriate for the first cycle operation and used this as a basis for its accident evaluation.

2.0 FUEL DENSIFICATION AND ITS EFFECTS

2.1 General Discussion

2.1.1 Fuel Densification Mechanism

The fuel densification recently observed in operating reactors is believed to be an irradiation induced process as compared with the thermally induced densification that takes place at high temperatures. In the latter case the densification is the result of temperature activated processes and steep thermal gradients in UO_2 fuel pellets. Pores initially present in the fuel migrate up the thermal gradients and a denser crystalline structure is formed.

Thermally-induced densification has been observed to occur as a result of in-reactor sintering, but this process requires temperatures of about 2400°F to 2700°F, and a structurally metastable fuel.(1) A structurally metastable fuel is one composed of pellets in which the sintering process was interrupted before it had gone to completion, or was carried out at too low a temperature for the times involved to complete the sintering process.

The structural changes in the recently observed fuel densification occurred in a relatively low temperature region of 700°F to 1900°F in the fuel. In typical operating light water reactors almost all of the fuel operates in this temperature range. Thus most of the fuel densification experienced in reactors was at temperatures where little or no thermal restructuring occurred and where thermally-activated processes were so slow as to be insignificant. Examinations of metallographic cross sections of irradiated fuel and immersion density measurements of irradiated fuel operated at these low temperatures confirm that the pellet densification occurs by annihilation of pores.[1]

The observed densification under these reactor operating conditions is believed to be an irradiation induced vacancy diffusion process. This diffusion of vacancies, provided by fission events, causes the vacancies to migrate towards the grain boundaries, free surfaces and dislocations in the pellets thus densifying the fuel.

Estimates by the staff suggest that, for fission rates of interest for light water power reactors (3 to 5 x 10^{13} fissions/cm^3sec), an in-pile diffusivity for uranium is obtained for the low temperature region that is roughly equivalent to an out-of-pile thermally activated diffusivity corresponding to a temperature of 2550^{o}F. This temperature is only slight below those used in many UO_2 sintering processes.

Examination of density changes in irradiated fuel has shown that, for operating times of less than 14 hours, temperature independent densification has not occurred, but that after approximately 2000 hours of operation fuel densification probably has been completed. The examinations also indicate that the densification is not isotropic, but more predominant in the axial direction. The evaluation model for the effects of fuel densification specified by the staff(1) requires the conservative assumption of instantaneous and anisotropic fuel densification.

2.1.2 Effects of Fuel Densification

Densification of fuel causes a decrease in the volume of the fuel pellet with corresponding changes in the pellet radius and length. There are three principal effects associated with fuel densification:

(a) The decrease in the pellet length will cause the linear heat generation rate to increase in direct proportion to the decrease in pellet length.

(b) The decrease in the pellet length can lead to the formation of axial gaps within the fuel column due to pellet hang-up, resulting in an increased local thermal neutron flux and the generation of local power spikes in the fuel pin containing the gap and in adjacent fuel pins.

(c) The decrease in the pellet radius increases the radial clearance between the fuel pellet and fuel cladding causing a decrease in the gap thermal conductance, and consequently in the capability to transfer heat across the radial gap. This decrease in heat transfer capability will cause the stored energy in the fuel pellet to increase. The effects of the reduced radial gap conductance become more pronounced during various transient and accident conditions.

In summary, the effects of fuel densification cause an increase in the linear heat generation rate of the pellet, create the potential for a local power spike in any fuel rod, decrease the heat transfer capability from the fuel rod and cause the fuel rod to contain more stored energy. To assess the safety implications of fuel densification, these effects have been evaluated by B&W for their current production fuel under various modes of reactor operation.

2.1.3 Manufacturing Parameters

The properties and dimensions of B&W UO_2 pellets, fuel rods and fuel rod assemblies are described in detail in the FSARs and individual fuel densification reports for each facility. The B&W fuel rods considered in the staff's evaluation are of the prepressurized type as contrasted to unpressurized fuel rods. Prepressurized fuel rods have less of a tendency for cladding collapse than unpressurized fuel rods.

The effects of fuel densification depend on the as-fabricated properties of the fuel which in turn are dependent on the control of many variables in the manufacturing processes of the UO_2 pellets (including the complete conversion cycle for the UO_2 raw material), of the Zircaloy 4 clad tubing and of all other components for a fuel assembly, and on the assembly process of the individual components into a completed fuel element bundle. The staff reviewed the following steps in the manufacturing and assembly process for the B&W fuel:

(a) preparation of UO_2 powder by the ammonium diuranate (ADU) method,

(b) fabrication of dished cylindrical pellets by cold pressing and sintering techniques,

(c) finishing of sintered pellets to final cylindrical dimensions,

(d) characterization and inspection of pellets for physical and chemical properties,

(e) procurement or production and inspection of Zircaloy clad tubing and all other component parts for an assembly,

(f) assembly of fuel element from component parts,

(g) prepressurization of fuel element with helium,

(h) inspection procedures for fuel element acceptance,

(i) incorporations of fuel elements into a 15 x 15 fuel element assembly, and

(j) inspection procedures for fuel element assembly acceptance.

The above described processes are conventional for the nuclear fuel industry. Based on the staff review of the production techniques and the quality control applied thereto the staff does not expect these factors to create any new or unusual densification effects for the B&W fuel.

2.1.4 Operating Parameters

Operating parameters related to the effects of fuel densification include, among others, the initial density, peak power, burnup, fission rate, and internal gas pressure. The effects of these parameters and their interrelationship have not been clearly established at this time, primarily due to the lack of sufficient experimental and operating data within the reactor community. However, some limited and preliminary conclusions can be drawn from available data.(1)

Fuel densification (fuel column shrinkage) decreases as the initial fuel density is increased. In order to reduce the potential effects of in-reactor fuel densification, B&W investigated experimentally the resintering effects on typical fuel pellets received from its vendors. The staff reviewed the B&W proprietary information[4] and concluded that the initial density was increased by resintering. The data also indicate that densification is more isotropic than prescribed in the staff's fuel densification report which states that the fractional change in pellet length should be assumed to be 1/2 of the fractional volume change and the fractional change in radius to be 1/3 of the fractional volume change. However, the B&W experiments were performed without any axial loading forces; in addition, the observed densification was thermally induced, i.e, without irradiation as under reactor operating conditions.

2.2 Mechanical Integrity of Cladding

Clad creepdown and time-to-collapse are two phenomena that affect the mechanical integrity of fuel cladding. Although similar in terminology and concept they differ both in methodology and in their effect on the behavior of the fuel.

2.2.1 Clad Creepdown

Clad creepdown is the term used to indicate the phenomenon which affects the geometry of the gap between the fuel pellets and the cladding. Clad creepdown causes a reduction in the gap size and thus results in an increased gap conductance.

A free-standing cylindrical tube when subjected to elevated temperature and a net external pressure undergoes a change in diameter due to creep. Normal production tubes are not perfectly cylindrical and have an as-fabricated ovality tolerance. Under high external pressure the tube will deform in an oval mode forming major and minor axes. The minor axis will touch the fuel first and, from them on, only the major axis will undergo creep. This overall behavior is known as clad creepdown.

The effects of clad creepdown have not been considered by B&W. This assumption is realistic in predicting gap conductance at beginning of life conditions (BOL) since no creep has occurred. At later times in life the assumption of no clad creepdown results in a conservative gap closure rate and attendant conservative estimate in gap conductance.

As discussed in Section 2.3.2 the elastic loading due to the difference between the plant system pressure and the fuel pin internal pressure are included in the B&W evaluation.

2.2.2 Time-To-Collapse

Time-to-collapse is time required for an unsupported cladding tube to flatten into the axial gap caused by fuel densification.

The accident analyses have been based on the assumption of uncollapsed cladding (see Section 1.2). The staff has reviewed the B&W model used to calculate the time at which fuel rod cladding collapse would be expected. The staff also performed independent calculations using the computer code BUCKLE.[6] The BUCKLE code calculates the creep collapse time of an initially out-of-round tube subjected to a net external pressure under high temperature and irradiation exposure.

The staff's calculations indicate that the time-to-collapse exceeds the proposed one cycle of operation with adequate margin. The Point Beach 1 and H. B. Robinson reactors using the Westinghouse prepressurized fuel design with comparable cladding parameters as the B&W fuel, have experienced approximately 13,000 hours of operation without creep collapse. The staff, therefore, concurs that fuel rod collapse is not expected for B&W fuel during the first cycle of operation for those reactors identified in Section 1.1. However, the staff has informed B&W[7] that an acceptable model for time-to-collapse calculations is necessary for subsequent fuel cycles.

The staff is not aware of any observation of cladding collapse experienced by a prepressurized tube.

2.3 Gap Conductance

2.3.1 General

The transfer of heat from the fuel pellet to the cladding is a function of the thermal conductance across the radial gap between the pellet and the cladding. The effect of fuel densification is to increase the size of the radial gap, which causes a decrease of the gap conductance and results in an increase of stored energy and temperature in the fuel pellet. B&W's calculation of the gap conductance used in analyzing the behavior of the fuel for all modes of reactor operation, is based on the staff guidelines,[(1)] which require the assumption of instantaneous and anisotropic fuel densification of the pellet.

The staff further requires[(1)] that until suitable models for clad creepdown are developed and verified, gap conductance should be evaluated with the assumption that clad creep does not reduce the gap size leading to gap closure. As discussed previously, B&W does not consider creepdown in the gap conductance calculation.

The gap conductance is, a function of gap size, the amount and composition of the gas in the gap (initial fill gas, fission gas, sorbed gas), surface roughness of the fuel and clad, the material properties of the fuel and clad, the contact pressure in the case of fuel-clad contact, and the linear heat generation rate in the fuel. Although the gap conductance is an important factor in establishing the fuel pin stored energy, other factors such as fuel conductivity, surface effects, and flux depression factors must also be considered.

2.3.2 Evaluation of B&W Code TAFY

The B&W computer code TAFY(8) was used to calculate the gap conductance, fuel temperatures, and stored energy giving appropriate considerations to the effects of fuel densification. The code analyzes the transfer of heat generated in the fuel to the coolant outside the cladding by calculating the temperature profile, stored energy, and gap conductance between fuel pellet and clad.

The parameters and phenomena that determine the heat transfer from the fuel to the coolant as indicated in Section 2.3.1 are either input to the TAFY code or are incorporated as analytical models in the code. The staff reviewed and evaluated the various assumptions for the input and models as used in the TAFY code in its application to B&W fuel.

As stated in Section 2.3.1, B&W assumes instantaneous, and anisotropic fuel densification at BOL and no clad creepdown in accordance with the staff guidelines. However, elastic loading of the clad by reactor coolant system pressure is included in TAFY and reduces the diametral gap by approximately 0.5 mil for the fuel rod cladding. The decrease in gap size is approximately offset by the thermal expansion of the cladding.

The thermal expansion of the fuel is calculated in TAFY using the temperature difference between the volumetric average fuel temperature and the ambient temperature, and using the thermal expansion coefficient, α, based on the data of Conway, Fincel, and Hines[9] for the volumetric average fuel temperature. This method is a simplified but acceptable approach for calculating fuel thermal expansion.

Fuel swelling due to the formation and accumulation of fission products in the UO_2 fuel increases the fuel volume, thus decreasing the gap size. The fuel swelling calculations in TAFY are based on a volumetric increase, which is assumed to partially reduce the fuel porosity and to partially cause fuel swelling in the axial and radial direction of the pellet.[10] The staff concludes that the fuel swelling in TAFY is realistic and acceptable.

The flux depression factors for the UO_2 fuel pellets are determined from physics calculations and are used in TAFY to calculate the fuel stored energy. The staff concludes that the factors are acceptable.

The expression for the fuel thermal conductivity in TAFY is based on the data by Lyons,[11] which result in a thermal conductivity integral of 93 watt/cm. Appropriate corrections are made for variations in fuel density from the 95% theoretical density (TD) which was used by Lyons.

The gap conductance is also a function of the amount and chemical composition of the gas in the gap between fuel pellet and clad. The gas is composed of the initial fill gas, which is helium for the B&W fuel pins, the sorbed gas released from the fuel to the gap, and the fission gases produced and released from the fuel as a function of time. In TAFY the gap conductance is calculated with the conservative assumption that the entire sorbed gas is released at BOL, with the initial amount of sorbed gas being input to the code. The release of the fission gases is treated in TAFY as a function of volumetric average fuel temperature based on the data by Hoffman and Coplin.[12] The conductivity of the gas mixture is based on the evaluation of Holmes and Baernes.[13] The staff concludes that the models in TAFY for the release of sorbed and fission gases and for the gas conductivity are acceptable.

The gap conductance determined by TAFY is based not only on heat transfer by radiation between the fuel pellet and the clad (generally only a small contribution) and by conduction through the gas in the gap, but also is based on heat conduction at a solid to solid, partial contact area which is assumed to exist between the fuel pellet and cladding. The partial contact area, C_A, is calculated by the expression

$$C_A = 0.1 + 0.9 \times 0.1^{(100\ G/D)}$$

where G is the gap size and D is the fuel pellet outside diameter. From this expression it is noted that for any pellet diameter the minimum partial contact area, independent of gap size is $C_A = 0.1$, i.e., 10% of the pellet circumference is in contact with the clad. The applicant based the concept of a partial contact area on the evaluation by Kjaerheim and Rolstad.(14) As discussed in Section 2.3.3, these investigators measured fuel and clad temperatures of fuel pins with cold diametral gaps quite small (0.0018 - 0.006 inch) in comparison to the densified gap of B&W fuel.

The equation for the partial contact area was derived by Kjaerheim and Rolstad together with a UO_2 thermal conductivity equation and the constants were adjusted to adequately confirm their measured fuel centerline temperatures. A model for UO_2 thermal expansion was assumed along with a host of other input variables for their prediction. The application of the C_A factor was explained as a method to account for possible fuel cracking that could lead to fuel-clad contact.

As discussed in Section 2.3.3 the B&W code TAFY, with C_A included, consistently overpredicts the fuel temperatures measured by Kjaerheim and Rolstad, which were the basis for the derivation of C_A. However, the conservative overprediction does not exist when TAFY is used for comparison with the experimental data by Balfour, Christensen and Ferrari[(15)] which were obtained for relatively large gap sizes of 24.5 mil. Thus the use of C_A is questionable when disassociated from other features in the Kjaerheim - Rolstad model. The staff concludes that the partial contact term C_A, may be very useful for predicting temperatures of the experiment from which the model was derived but should be used with caution unless verified by comparison with independent data.

Restructuring of the fuel is included in TAFY and is based on restructing temperatures reported by MacEwan.[(16)] Columnar grain growth is assumed at a temperature of 3200°F based on the BOL temperature distribution in the fuel. The porosity released in the restructuring process is assumed to migrate toward the center of the fuel leading to the formation of a center void and resulting in a density of 100% TD for the restructured fuel. The result of this center void is a reduction in maximum fuel temperature and thus a lower stored energy in the fuel pin. Photomicrographs of cross sections of fuel pellets after high burnup, provided by B&W show the existence of center voids and of restructured fuel due to columnar grain growth. Detailed information on the pellet (initial density, enrichment and pellet dimensions) and on the power history of the pellet during exposure were not available.

The staff concludes that although restructuring of the fuel due to columnar grain growth can take place for certain power histories and operating conditions, insufficient information was provided by B&W to establish a temperature of 3200°F as the temperature at which columnar grain growth is initiated, and to attribute a density of 100% TD to the restructured fuel. In addition fuel densification induced by radiation could be completed before the fuel experiences a temperature of 3200°F and thus foreclose the fuel restructuring due to columnar grain growth leading to the formation of a central void.

2.3.3 Comparison of TAFY Code with Experimental Data

The computed results from an analytical model should relate to the observations and measurements made during irradiation experiments or in post-irradiation examinations. Comparison of the analytical results with experimental measurements is an essential test of the analytical models of the phenomena being described and of the computer code which is based on these models. Since the heat transfer from the fuel pin to the reactor coolant outside the clad depends on numerous design and operating variables which are also input to a computer code, a complete evaluation of the code would require extensive parameter studies and comparisons with experimental data. However, the experimental information available is rather limited; most references are restricted to particular design and operating conditions with only the linear heat rate being used as the experimental variable. Furthermore, some of the variables used in a particular experiment are not reported by the investigators and appropriate best estimates must be made for the code input.

B&W has calculated gap conductances and fuel temperatures using the TAFY code for conditions corresponding to those reported in various references. The measured quantity in the experiments is either a fuel center line temperature measured with thermocouples or the fuel melt or grain restructuring radius measured in post-irradiation examinations. From these measurements fuel centerline temperature and/or gap conductance are calculated and reported. The staff has compared these parameters deduced from experiments with the corresponding TAFY predictions on the basis of the following ratios:

$$R_T = \frac{\text{code predicted temperature (°F)}}{\text{experimental temperature (°F)}}$$

$$R_h = \frac{\text{experimental gap conductance (Btu/hr-ft}^2\text{-°F)}}{\text{code predicted gap conductance (Btu/hr-ft}^2\text{-°F)}}$$

The code is conservative with respect to an experimental value if these ratios are greater than 1.0.

A comprehensive summary of the comparisons is listed in Table 2.1 and discussed below. The initial sorbed gas content, $S(cm^3/g)$, is listed as a parameter in the table. Since no value is given in any of the references for this parameter, its effect was evaluated parametrically with TAFY in some cases.

The data reported by Kjaerheim and Rolstad[(14)] were obtained for fuel pin geometries with cold diametral gaps ranging from 1.85

mil to 6.61 mil, linear heat rates ranging between approximately 2 to 15 Kw/ft and short irradiation times representing BOL conditions. A total of 82 reported experimental fuel temperatures were compared with TAFY predictions. With an assumed initial sorbed gas content of S = 0.05 all data are predicted conservatively, i.e., $R_T > 1.0$, with an average value is $\overline{R_T} = 1.21$. As discussed in Section 2.3.2, these data form the basis for the partial contact term, C_A, used in the TAFY code.

The experiments by Balfour, Christensen, and Ferrari[15] were performed to determine UO_2 thermal conductivity from fuel temperature measurements. Data for two capsules were obtained that differ in U^{235} enrichment for otherwise identical conditions: 24.5 mil diametral gap, linear heat rates from 2 to 26 Kw/ft, BOL conditions, and helium as initial fill gas at a pressure of 1 atm. The average ratio, $\overline{R_T}$, for the 18 data points considered decreases from 1.02 to 0.95 when the assumed initial sorbed gas content, S, is changed from 0.05 to 0.0. The comparison indicates only a small change in $\overline{R_T}$ with respect to S and the ratio of $\overline{R_T} = 0.95$ at S = 0 suggests that TAFY predicts the experiment reasonably well. However, as seen from Table 2.1, the data that are predicted conservatively by TAFY decreases from 14 (S = 0.05) to 6 (S = 0). The difference between measured and

TAFY predicted temperatures is shown in Figure 2.1, which also includes two data points at low linear heat rates that were not considered in Table 2.

B&W considers the data presented in references 14 and 15 appropriate for an evaluation of the TAFY code since fuel temperatures rather than deduced gap conductances can be compared and since the experiments were performed with cold diametral gaps (6.5 mil and 24.5 mil) which bound diametral gaps calculated for densified B&W fuel. B&W determined a combined average ratio of TAFY predicted to measured temperatures of $\overline{R_T}$ = 1.18, using S = 0.05 for the initial sorbed gas content. The staff concludes that the references present two distinctly different sets of data which therefore cannot be combined. The larger set of conservative data[(14)] overshadows the conclusions that can be drawn from the smaller set of data,[(15)] which exhibits a wide range of ratios. The use of a sorbed gas content of S = 0.05 is a reasonable estimate in the TAFY calculations for the above experiments in the absence of a measured value, but it is not conservative, particularly since a value of S = 0.02 had been used in the TAFY calculations for some B&W fuel.

Ditmore and Elkins[(17)] report gap conductances for a cold diametral gap of 15.8 mil and linear heat rates ranging between 16 and 21 kW/ft. The TAFY predicted gap conductances for corresponding conditions for the data reported result in average ratios, $\overline{R_h}$, of 1.72 and 1.20 for the assumed sorbed gas content of 0.05 to 0.0, respectively. However, for the 8 predictions only 5 were conservative at S = 0.05 and only 3 were conservative at S = 0.0. Figure 2.2 shows measured and TAFY predicted values and clearly indicates that the average ratio, $\overline{R_h}$, is strongly influenced in the conservative direction by two of the data points. The minimum ratio for this set of data with the assumption of S = 0.02 is R_h = 0.76.

A fourth set of data with a cold diametral gap comparable to the densified gaps of the B&W fuel is reported by Duncan:[18] 12.0 mil gap, 11-24 kW/ft linear heat rate, BOL conditions, 4 measurements. TAFY predicts the reported gap conductance conservatively in all cases and the average ratio is $\overline{R_h}$ = 1.97. Since extreme care was taken in outgassing the fuel prior to encapsulation in the clad, one can assume that no sorged gas was present, while B&W assumed a value of S = 0.01. However, it is expected that the TAFY predictions would also be conservative if an assumption of zero sorbed gas were made for the calculation.

2.3.4 Conclusions

The staff concludes that the B&W code TAFY in its present form should not be used to calculate gap conductance, fuel temperature and stored energy for the B&W fuel. The bases for this conclusion are: (1) The contact area term, C_A, is phenomenologically not fully understood and substantiated. (2) The effect of power history on fuel restructuring leading to the formation of a center void and the restructured density of 100% TD have not been fully established. (3) TAFY does not predict the experimental data sets in a consistent manner, in particular for variable gap size. The staff concludes that on an interim basis these deficiencies in the TAFY code can be accounted for in the fuel pin thermal analyses by (1) reducing the TAFY calculated gap conductance by 25% (all data of ref. 17 with S = 0.02 will be predicted conservatively) with a corresponding increase in fuel temperature and (2) by taking no credit for fuel restructuring and the resulting center void.

2.4 Fuel Pin Thermal Analysis for B&W fuel

To compare B&W's TAFY code to the thermal predictions of the staff's GAPCON[19] code, the specific first cycle fuel characteristics of Duke Power Company's Oconee 1 (the prototype design B&W reactor) were used. First cycle fuel characteristics from other B&W designed reactors are similar to those for Oconee 1.

B&W calculated gap conductance, fuel temperature and stored energy for the Oconee 1 fuel using the B&W code TAFY with the assumptions and models discussed in Section 2.3.2. Other pertinent input data for the calculation and the results for a linear heat generation rate of 16 Kw/ft at BOL conditions are listed in Table 2.2 as the design case. The contact area is approximately 13% and its contribution to the total gap conductance is approximately 35%. The TAFY calculated gap conductance without fuel pellet to clad contact (C_A=0) would be approximately 75% of that with C_A=0.13 listed in Table 2.2 as the design case.

The staff calculated gap conductance and fuel temperature for identical conditions using the computer code GAPCON,[19] a code that is presently under development by the staff in cooperation with its consultant, Pacific Northwest Laboratories. The results are also listed in Table 2.2, which shows that for the design case the TAFY and GAPCON calculated values for gap-conductance (But/hr-ft^2-°F) and maximum fuel temperature (°F) are 1052 vs. 697 and 4000 vs. 4517 respectively. The design case uses as input a fuel pellet diameter of [ ] inch, calculated according to the staff guidelines (instantaneous densification with a 2σ lower limit), a sorbed gas content of 0.02 cm^3/g which is the Oconee 1 nominal specified value for the upper tolerance limit, and a clad inside diameter of [ ] inch, based on the mean as-built clad diameter combined with the standard deviation for the clad ID and the fuel pellet OD.

In order to partially compensate for the derating of the TAFY code as discussed in Section 2.3.4 B&W used as-built data

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

for the sorbed gas content [ ] and the clad ID [ ] instead of the conservative data used in the analysis of the design case. The staff concludes that the as-built data are appropriate. The results of TAFY and GAPCON calculations for the as-built data are listed in Table 2.2. With the indicated input change the TAFY gap conductance increases from 1052 to 1100 Btu/hr-ft^2-°F which is then reduced by 25% to 825 Btu/hr-ft^2-°F. The maximum fuel temperature increases from 4000 to 4320°F, due to the reduction in gap conductance and the effect of no restructuring. The corresponding values calculated by the staff are 775 Btu/hr-ft^2-°F and 4466°F. The staff concludes that B&W fuel pin thermal analysis using the TAFY code, derated according to the staffs evaluation (Section 2.3.4), is acceptable. The calculated gap conductance and maximum fuel temperature are used to determine the maximum linear heat rate (Kw/ft) for the fuel on the basis of fuel melt and ECCS criteria.

Table 2.1 Comparison of TAFY Predicted and Experimental Fuel Temperatures and Gap Conductances

Reference		HPR-80 (14)	WCAP-2923 (15)	CVNA-142 (18)	GE-AEC (17)
gap, cold diametral (mil)		1.85-6.61	24.5	12.0	12.8
linear heat rate (Kw/ft)		2-15	13-25	11-24	16-21
comparison parameter		T	T	h_{gap}	h_{gap}
data in set		82	18	4	8
S = 0	$\overline{R}$	-	0.95	-	1.20
	R max	-	1.02	-	3.33
	R min	-	0.71	-	0.58
	R >1.0	-	6	-	3
S = 0.01	$\overline{R}$	-	-	1.97	-
	R max	-	-	2.27	-
	R min	-	-	1.17	-
	R >1.0	-	-	4	-
S = 0.02	$\overline{R}$	-	1.00	-	1.54
	R max	-	1.07	-	4.23
	R min	-	0.79	-	0.76
	R >1.0	-	11	-	4
S = 0.05	$\overline{R}$	1.21	1.02	-	1.72
	R max	1.38	1.08	-	4.74
	R min	1.08	0.83	-	0.84
	R >1.0	82	14	-	5

$$R_T = \frac{\text{TAFY predicted temp. (°F)}}{\text{Experimental temp. (°F)}}$$

$$R_T = \frac{\text{experimental gap conductance (Btu/hr-ft}^2\text{-°F)}}{\text{TAFY predicted gap conductance (Btu/hr-ft}^2\text{-°F)}}$$

Table 2.2 Oconee 1 Fuel Pin Thermal Analysis (16 Kw/ft, BOL, 96.5% TD)

Code Condition	TAFY Design	TAFY As-Built	GAPCON Design	GAPCON As-Built
Clad ID (in)	.3780	[ ]	.3780	[ ]
Fuel OD (in)	[ ]	[ ]	.3652	[ ]
Cold Gap (in)	.01278	[ ]	.0128	[ ]
Sorbed Gas (cm^3/g)	.02	[ ]	.02	[ ]
Restructure	yes	no	no	no
Contact area, C_A	.013	-	0	0
h_{gap} (Btu/hr-ft^2°F)	684	-	697	775
h_{C_A}	368	-	0	0
h_{total}	1052	1100	697	775
.75 h_{total}	789	825	-	-
T_{max}(°F)	4000	4320	4517	4466

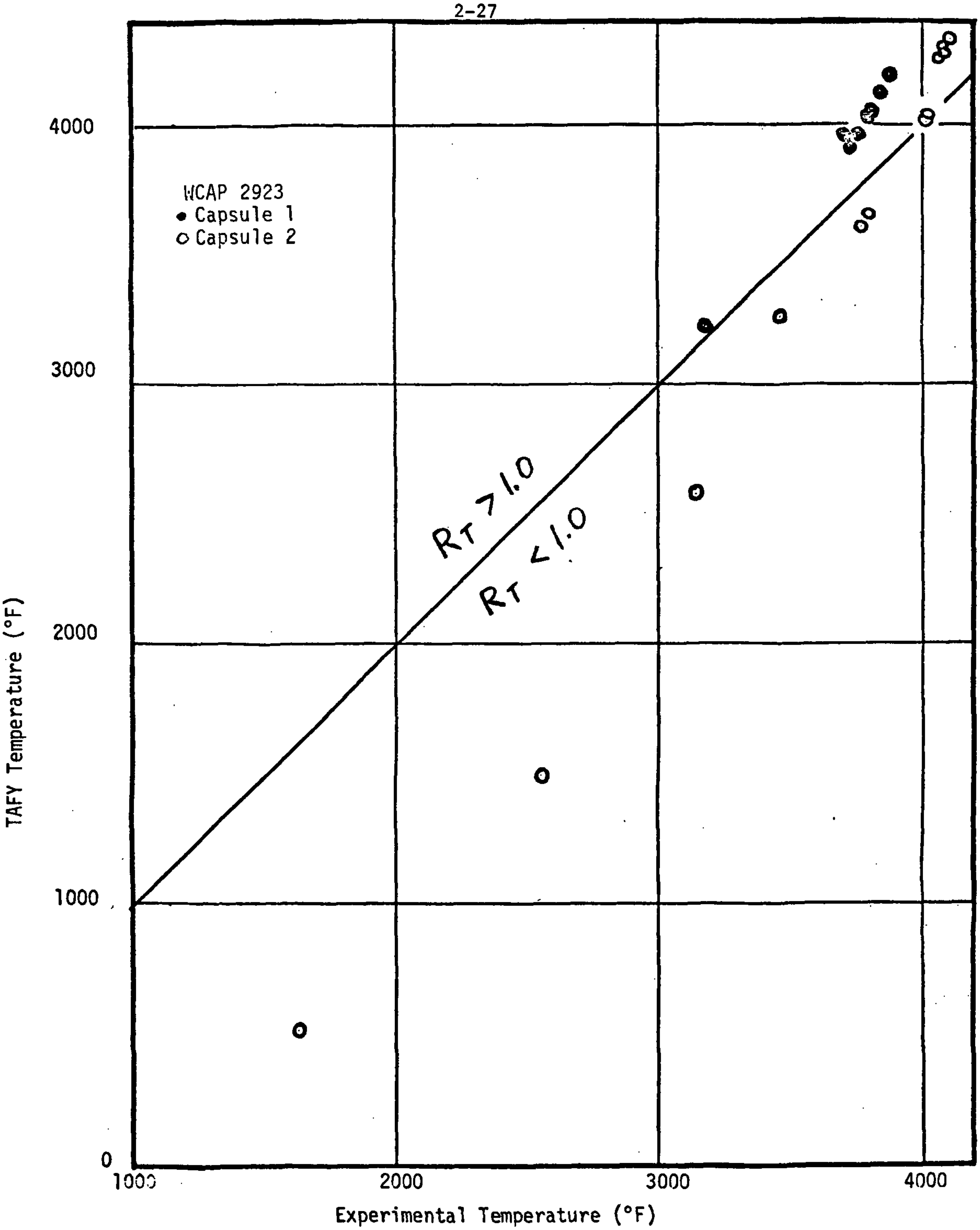

Figure 2.1 Fuel Temperatures - TAFY Predicted and Experimental (WCAP-2923)

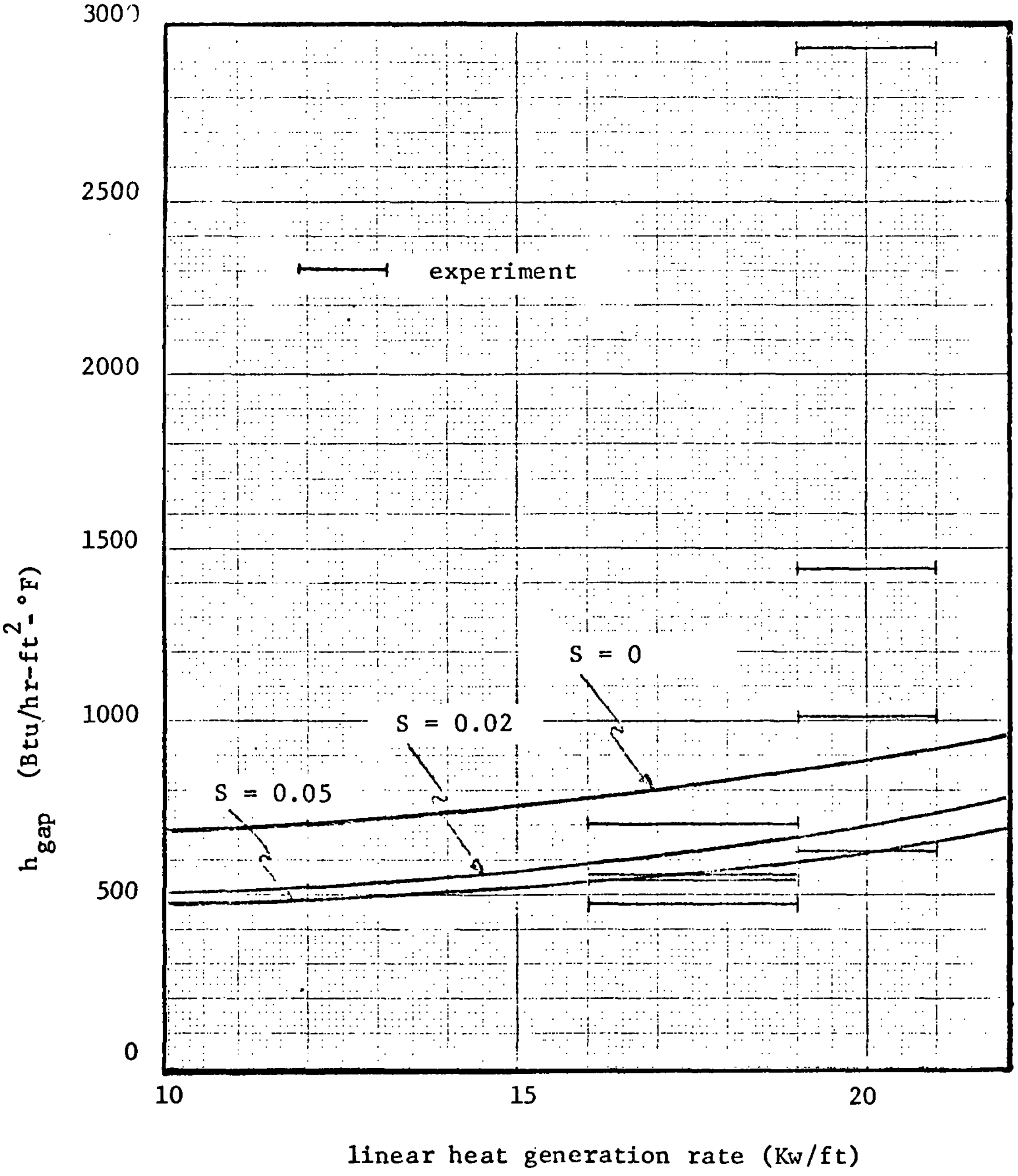

Figure 2.2 Gap Conductance - TAFY Predicted and Experimental Variable Sorbed Gas (NEDM 10735, Rod AEG)

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.

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 and the probability of gap size was changed to conform to that recommended by the staff.

The calculations by B&W 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 at the top of the core. The overall calculation falls within the range examined 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 linear heat generation rate due to the reduced fuel column height based on the instantaneous densification from the minimum initial density to a final density of 0.965 TD in accordance with Section 4.5 of reference 1. The reactor operating limits, which are made part of the Technical Specifications for each facility, are based on thermal and ECCS criteria (minimum DNBR of 1.3 and no fuel melting) which restrict the maximum linear heat generation rate through the reactor power vs axial offset correlation.

3.2 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 B&W and the staff for the prototype reactor, Oconee 1. 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, B&W 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 individual facility FSARs. 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.

The staff has also evaluated some of the thermal hydraulic analyses associated with steady state operation and the loss of reactor coolant flow transient (LOF). The evaluations were performed using the computer program COBRA III-C.[20] This program calculates the heat transfer and fluid flow conditions in rod bundle nuclear fuel element subchannels during both steady state and transient conditions. It uses a mathematical model that considers both turbulent and diversion cross flow mixing between adjacent subchannels. The thermal model considers radial conduction within the fuel. Axial and circumferential conduction are ignored. The model uses circumferential averaged coolant temperature and surface heat transfer coefficients either input or calculated from the subchannel hydraulic data at each axial node. The thermal properties of the fuel and clad are considered constant and uniform throughout a transient calculation. Gap conductance is input to the calculations, and is assumed to be constant throughout the transient.

To compare B&W's thermal-hydraulic code predictions to the staff's COBRA-IIIC predictions, the specific first cycle fuel characteristics of Duke Power Company's Oconee 1 (the prototype of B&W designed reactors) were used. First cycle fuel charcteristics from other B&W designed reactors are similar to those for Oconee 1. The staff evaluations were performed for a rod bundle consisting of 15 rods (12 fuel rods and 3 control rod guide tubes), which is part of the hottest assembly in the core. Identification of subchannels and rods are shown in Figure 3.1. The following conditions apply to the events analyzed:

a. Reduction of the clad OD due to creep and its associated effect on gap conductance is ignored, conservatively.

b. Fuel densification is computed from an initial density of [ ] TD (minus 2σ) and a final density of 0.965 TD. The average active height of the fuel in the core is [ ], which accounts for densification but not for thermal expansion of the fuel. Thermal expansion of the fuel would tend to lower the heat flux and is conservatively neglected.

c. Coolant inlet flow to the hot subchannel (channel #10) is reduced to 95% of the hot assembly average flow and flow to all other subchannels of the bundle (Figure 3.1) is reduced to 99%.

d. Core wide radial flux factor for hot assembly is $F^R = 1.68$.

e. Local radial flux factors (ratio of peak rod power to average assembly power, F^R local) for each rod in the bundle are applied. F^R local (hot rod #11) = 1.061 (without densification).

f. An engineering hot channel factor* of $F_Q^E = 1.011$ is applied to the enthalpy rise in the hot channel, #10. Another engineering hot channel factor* of $F_Q^{E}{}'' = 1.014$ is applied to the surface heat flux of rod #11. The location of the hot spot for steady state conditions is the axial position with the minimum DNBR (Z = 98 inches). This is determined by placing the spike at the point of minimum DNBR for the densified length of [ ].

*The engineering hot channel factors, F_Q^E & $F_Q^{E}{}''$, are used to describe variations of fuel loading, fuel and clad dimensions, and flow channel geometry from perfect physical qualities and dimensions.

g. The axial flux factor is $F_Q^Z = 1.50$, based on a symmetric distribution.

h. A flux spike of 9.6 percent (see Section 3.1) is superimposed on the cosine distribution of all rods in the bundle at the axial location of minimum DNBR (see item e) resulting from an axial fuel gap at this location due to fuel densification. This 9.6 percent spike is extended over approximately 4 inches (2 axial nodes).

i. The fuel thermal model included: 6 radial nodes, 72 axial nodes, fuel density 641.1 lb/ft^3, fuel conductivity 1.65 Btu/hr-ft-°F, clad density 407 lb/ft^3, clad conductivity 9.3 Btu/hr-ft-°F, clad specific heat 0.12 Btu/lb-°F.

j. DNBR was computed by the W-3 correlation.[21]

The steady state calculations considered the effect of turbulent mixing parametrically by using a turbulent mixing coefficient, beta, of 0.011 and 0.02. A beta of 0.011 in COBRA gives closer agreement with B&W's code which uses an equivalent value of 0.02. This is probably because B&W's code does not consider diversion cross flow. In any case, the difference caused by either a beta of 0.011 or 0.02 is small, about 2 percent in DNBR. The loss of flow transient and the locked rotor accident (Section 4.2) were analyzed by the staff using the more conservative value of beta, 0.011.

Since the mechanical condition of the eight reactor internals vent valves is not directly monitored (i.e., open or closed), except for the Rancho Seco plant, and there has been limited operational experience with these devices, one vent valve was arbitrarily assumed not to be present in these analyses. This "open" vent valve permits bypass of about 5 percent of the reactor coolant flow, and reduces the DNBR margin for both steady state and accident condition by about 6 percent. The results of the staff thermal-hydraulic evaluation of the steady state condition with various assumptions are listed in Table 3.1.

In addition to the reference design power shape described above (F_Q^Z = 1.50 and F_Q^R = 1.68) another power shape which B&W used in their LOCA calculations was examined. The LOCA power shape (F_Q^Z = 1.786 at [ ] inches) produced a higher peak linear heat rate (17.4 KW/ft vs 15.6 KW/ft for the undensified condition at 102% power) than the reference design shape. This LOCA power shape was not used by B&W in their thermal-hydraulic analysis. B&W maintained that the reference design power shape was more conservative for thermal-hydraulic analyses. The COBRA steady state results at 114% overpower confirm that the reference design shape would produce a lower DNBR than the LOCA power shape and, therefore, is conservative for use in thermal-hydraulic calculations.

B&W also reanalyzed the loss of flow transient that would result from a loss of electrical power to the primary 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 1.56.

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.

The B&W analysis employed a digital computer code which considers six delayed neutron groups, control rod worths, rod insertion characteristics and trip (pump monitor) delay time to calculate the neutron power transient. An analog model was used to determine the flow coastdown rates. The flow coastdown was measured in hot functional tests at Oconee 1 and agreed well with these analytical predictions. These values of neutron power and flow rate were then used to calculate the DNBR as a function of time. This code is a two-channel model of the reactor core, one average channel and one hot channel.

The staff also conducted an independent calculation of the loss-of-flow transient. The transient DNBR predicted by COBRA for the LOF event are shown in Figure 3.2. The appropriate flow reduction and power decay curves from the FSAR and BAW-1387 Revision 1 were used. The minimum DNBR for the LOF was reached at 1.4 seconds, and is 1.66. By comparison B&W computed a minimum DNBR of 1.56 at about 1.5 seconds. Table 3.1 presents input data and results for the various COBRA steady state conditions and the loss-of-flow transient.

The COBRA III-C code contains the basic features required for a comparative evaluation of B&W's computations. We conclude that the reasonable agreement between the two sets of computations consitutes an acceptable audit of B&W's thermal-hydraulic calculations for steady state and LOF.

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

3.4 Conclusions

The models used by B&W to evaluate the effects of fuel densification on steady state and transient operation have been reviewed and evaluated by the staff.

The staff concludes on the basis of its review that the potential effects of fuel densification on the steady state and postulated transient operation can be evaluated in an appropriate manner by using the B&W evaluation models discussed above.

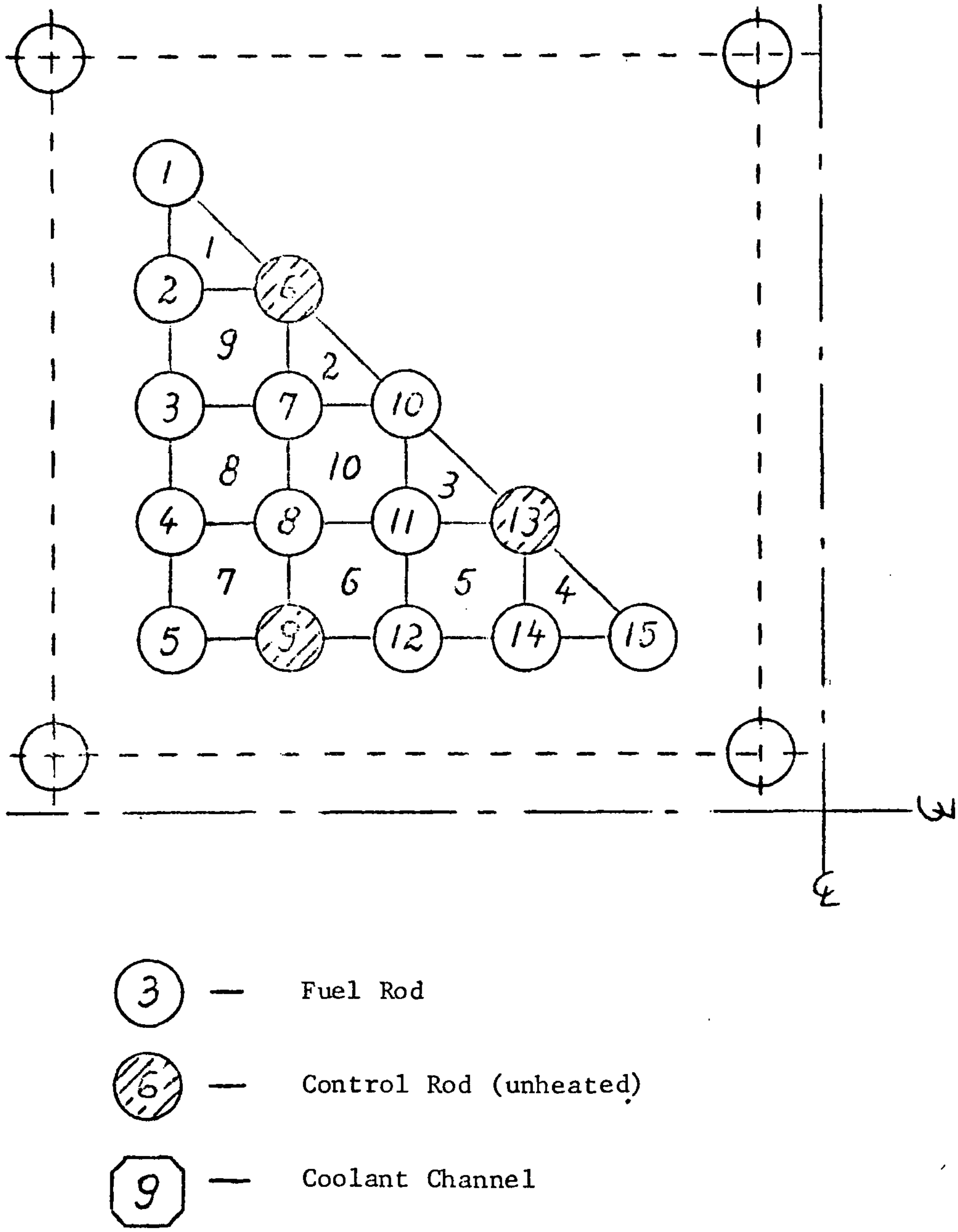

Figure 3.1 Location and Identification of Rod and Channel Geometry in Fuel Assembly As Used in COBRA IIIc Analyses for Oconee 1.

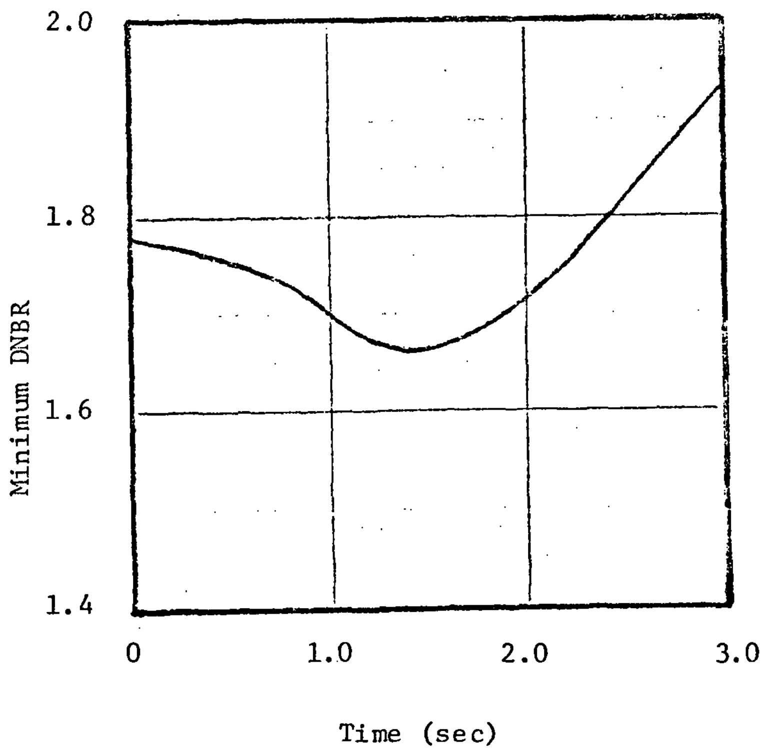

Figure 3.2 Minimum DNBR for Loss of Flow Transient as Calculated by COBRA III-C for Oconee 1

Table 3.1 Results of COBRA III C Calculations for Various Conditions of Steady State and Loss of Flow Transients for Oconee 1

Condition	Steady State(1)	Steady State(1)	Steady State(2)	Steady State(1)	Steady State(1)	Steady State(1,3)	4-Pump LOF(1,3)
Power (% nominal)	114	114	114	114	114	114	102
Turbulent Mix (beta)	0.02	0.011	0.011	0.02	0.011	0.02	0.011
Power Spike (%) @ 98 in	0	0	0	9.6	9.6	9.6	9.6
Inlet Temp.(4) (°F)	552.6	552.6	552.6	552.6	552.6	552.6	555.4
h_{gap} (Btu/hr-ft^2-°F)	500	500	500	500	500	500	420
DNBR min	1.57	1.54	1.67	1.54	1.51	1.45	1.66
time (sec)	-	-	-	-	-	-	1.4

(1) 1.50 symmetric axial power shape with 1.68 radial peaking factor

(2) 1.786 axial power peak at [ ] with 1.57 radial peaking factor

(3) vent valve assumed open

(4) Inlet temperature includes a +2°F uncertainty factor. System pressure assumed as 2135 psia (2200 psia nominal minus a 65 psia uncertainty).

4.0 Accident Analyses

4.1 General

Analyses of the consequences of various postulated accidents were presented in the individual facility FSARs.

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 B&W 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 2.0 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 are site related and therefore are reported separately for each facility. 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 are based on an assumed quantity of gas in the tank. The assumed quantity is consistent with the Technical Specification limits on maximum permitted reactor coolant system activity. Fuel densification will not affect the consequences of this accident.

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 at 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 1.0), 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 Analysis

The reactor coolant system for B&W designed reactors 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 facility, FSARs. 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 the reactor coolant would increase, causing fluid expansion with a resultant pressure transient which would reach a peak of approximately 15 psi above nominal.

The thermal analysis of the hot rod in the core was performed using design conditions with respect to power (102%), flow (95%), core inlet water temperature (nominal +2°F) and system pressure (nominal -65 psia). Following the onset of DNB (defined as DNBR <1.3), the Bishop-Sandberg-Tong flow film boiling correlation,[22] was used to predict heat transfer at the affected location. It is conservative to assume DNB at a DNBR of 1.3 since statistically there is a 95% confidence that 95% of the fuel pins are still in nucleate boiling.

The staff has reviewed the data from which the Bishop-Sandberg Tong correlation was derived. For its specific applicability to the analysis of the locked rotor accident for B&W reactors, the range of data obtained with regard to heat flux, mass flow rate and pressure was appropriate for use in the analysis. The staff concludes that the use of the Bishop-Sandberg-Tong flow film boiling heat transfer correlation is acceptable for use in the analysis of the locked rotor accident.

The staff calculated maximum cladding temperatures for the locked rotor accident. The subchannel arrangement and densification input were the same as discussed in Section 3.2 for the loss-of-flow transient. The power vs time and reactor coolant flow vs time were taken from the FSAR for input to the COBRA-III-C calculation. Whenever the DNBR decreased below 1.3 the film boiling computations were performed. The surface heat transfer coefficient was then decreased from a value of 5,000 Btu/hr-ft^2-°F to the value calculated by the Bishop-Sandberg-Tong flow film boiling correlation. No provision was included to allow recovery from DNB. For one case the nucleate boiling correlation by Thom(22) was used together with Tong's transition film boiling correlation.(22)

Four cases are presented in Table 4.2 and Figure 4.1 showing the effect of various conservative assumptions on the consequences of the locked rotor accident. The following parameters were varied: the gap conductance, the cladding collapse during the accident (which increases the gap conductance), and the heat transfer regime assumed. Case 1 in Table 4.1 is the most conservative combination of all parameters: a low initial gap conductance (420 Btu/hr ft °F) is assumed; the cladding is assumed to collapse onto the hot fuel pellets when the peak cladding temperature exceeds 1100°F thus transferring stored heat to the cladding; and fully developed flow film boiling is assumed whenever the DNBR drops below 1.3. This case results in a peak cladding temperature of 1720°F at 4.2 seconds. Case 2 is similar to Case 1 except that cladding collapse is not assumed. This case resulted in a peak cladding temperature of 1520°F. A constant gap conductance of 825 Btu/hr ft^2°F (the approximate value used by B&W for the Oconee 1 evaluation) is assumed for Case 3. In this case the DNBR drops to just slightly above 1.3 at 1.6 seconds and then recovers as the power decreases. Since flow film boiling was assumed to occur below a DNBR of 1.3, the cladding temperature did not significantly increase. The fourth case is similar to cases 1 & 2 in that a gap conductance of 420 Btu/hr ft^2°F is assumed, however, nucleate boiling heat transfer is properly considered in place of the forced convection coefficient, 5000 Btu/hr-ft^2-°F. A transition nucleate boiling term is also included in the flow film boiling calculation. The peak cladding temperature computed for Case 4 is 670°F. This case is considered to be the most realistic of the four cases in terms of cladding temperature but yet still conservative in terms of gap conductance and when film boiling occurs. B&W computed a maximum cladding temperature of 1300°F at 4.4 seconds. B&W assumptions included fully developed flow film boiling at a DNBR of 1.3 and a gap conductance of 825 Btu/hr-ft^2-°F.

4.3 LOCA Analysis

The B&W evaluation model described in the AEC Interim Acceptance Criteria and Amendments for Emergency Core Cooling Systems was used to evaluate the loss-of-coolant accident (LOCA). 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 LOCA analysis without the assumption of fuel densification is reported in the individual facility FSARs based on the 8.55 ft^2 split break in the cold leg at the pump discharge as the limiting break size and location.[23]

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, B&W 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. Reduced gap conductance during this time would be a benefit in that the rate of decay heat transferred across the gap to the cladding would be reduced. However, the benefit is not significant since the gap conductance is much larger than the film coefficient during the reflood period and hence is limiting with regard to heat transfer and cladding temperature.

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, B&W 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^oF. A hot channel factor of F_{HC} = 1.014 was used in the calculations. The radial peaking factor, F^R, including an uncertainty factor of 1.05 was varied until the calculated maximum cladding temperature approached the 2300^oF limit. Using the gap conductance as calculated with the TAFY code described in Section 2.3.2 a clad temperature equal or below 2300^oF is reached with a certain maximum linear heat rate. In order to accommodate a possible quadrant tilt of 5% during this design basis transient the allowable heat rate is further reduced by a factor of about 10% to account for uncertainties. This maximum allowable linear heat rate will be controlled by a control rod operating band between a rod index range specified in the Technical Specifications for each reactor.

4.4 Rod Ejection Accident

The control rod ejection transient has been reanalyzed by B&W 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 due 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 individual facility FSARs. Our 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 re-evaluation, thereby adding additional conservatism to the calculations. The maximum Technical Speficiation rod worth was used for the BOL calculations.

Our 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.

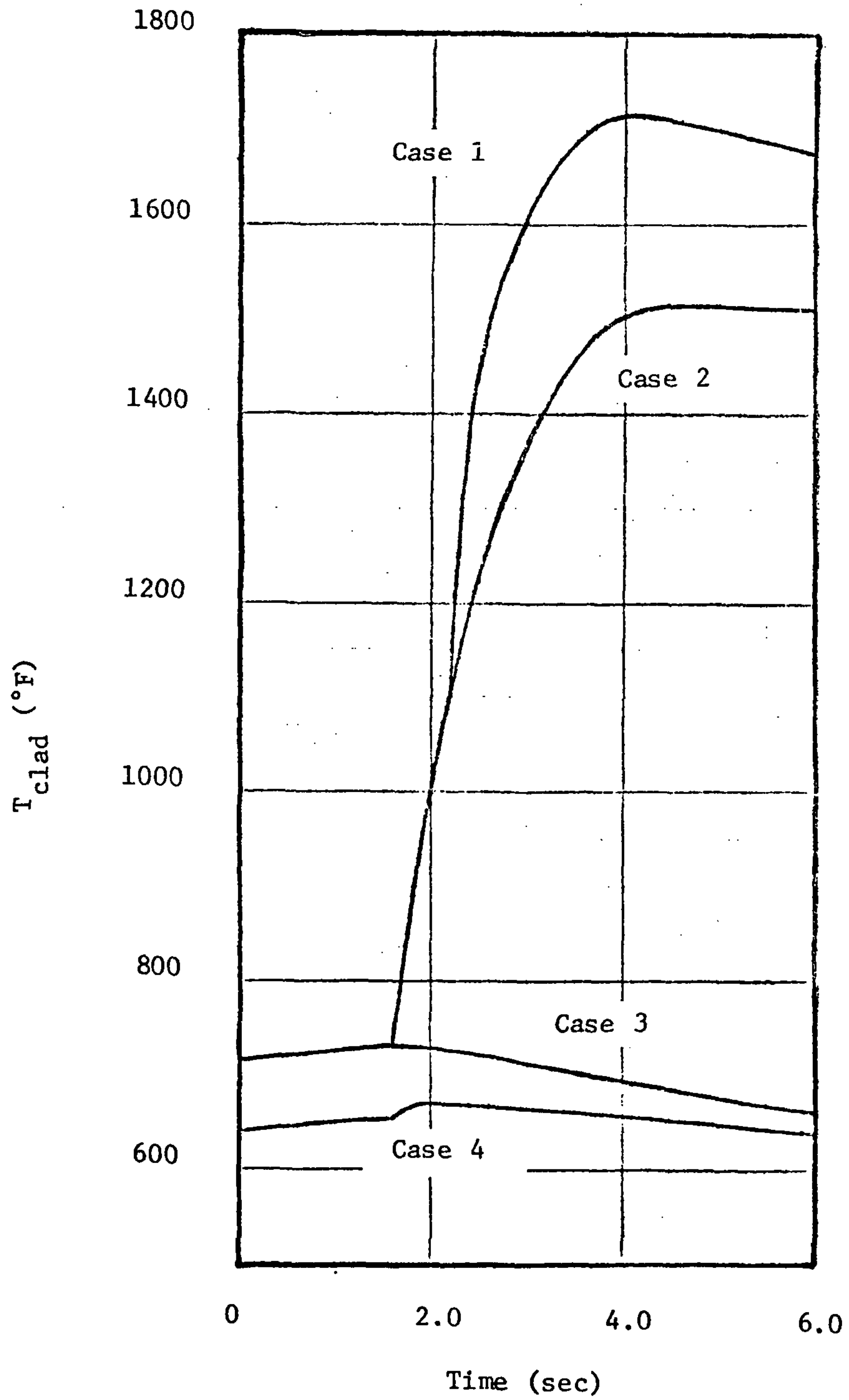

Figure 4.1 Maximum Clad Temperature for Locked Rotor Accident As Calculated by COBRA III-C for Oconee 1

Table 4.1 Results of COBRA III-C Calculations for Various Locked Rotor Conditions for Oconee 1

	Case 1	Case 2	Case 3	Case 4
Initial Power (% nominal)	102	102	102	102
Power Spike (%) at 98 in	9.6	9.6	9.6	9.6
Inlet Temperature (°F)	555.5	555.5	555.5	555.5
h_{gap} (Btu/hr-ft^2-°F)	420*	420	825	420
Heat Transfer Correlation	(1)	(1)	(1)	(2)
Max Clad Temp. (°F)	1720	1520	no increase	670
time (sec)	4.2	5.4		2.0

* increased to 10,000 Btu/hr-ft^2-°F when clad temperature > 1100°F

(1) forced convection (5,000 Btu/hr-ft^2-°F) changed to flow film boiling when DNBR < 1.3

(2) nucleate boiling changed to transient flow film boiling when DNBR < 1.3

5.0 SUMMARY AND CONCLUSIONS

The effects of fuel densification have been considered in analyses of normal operation, operation during transient conditions, and postulated accident conditions. On the basis of the staff review of the B&W calculations, and independent calculations performed by the staff and its consultants, the staff concludes that B&W's models, discussed in Sections 2.0, 3.0 and 4.0 adequately consider:

(1) The effects of densification during steady state and transient operation on the limits of DNBR, cladding strain, and centerline temperatures, such that they will not become less conservative than values previously established in the FSAR.

(2) The effects of densification are included in the calculation of fuel rod behavior during postulated loss-of-coolant accidents. The LOCA analysis is acceptable and complies with the June 1971 Interim Policy Statement.

(3) The Technical Specifications on individual facility will limit the fuel residence time to such a value as to assure no cladding collapse. Further, B&W does not include the creep down effect which tends to increase gap conductance with life time.

(4) The staff recommendations for calculating gap conductances and fuel temperatures (Section 2.3.4) as they are used in steady state, transient and accident conditions.

(5) Operating restrictions are necessary to assure compliance with considerations (1) through (3) above, and will be incorporated into the Technical Specifications.

(6) The effects of the individual facility "as built" fuel characteristics are included in the application of B&W's fuel densification models.

On the basis of the above considerations, the staff concludes that B&W's fuel densification models are in compliance with the staff's densification report;[1] and are acceptable for use in evaluating the effect of fuel densification.

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