DISTRIBUTION
Docket File
AEC PDR
DR Reading File
L Reading File
RP Reading File
Branch Reading Fil
JRBuchanan, ORNL
TBAbernathy, DTIE
EP A (3)
LMMuntzing
JFO'Leary
AGiambusso
JMHendrie
VStello
DJSkovholt
ACRS (16)
RO (3)
OGC
RJSchemel
RHVollmer
TJ
T
R
SV
ND
MJi
WOM
SKa
SATe

AUG 24 1973

Docket No. 50-219

Jersey Central Power & Light Company
ATTN: Mr. I. R. Finfrock, Jr.
Vice President - Generation
Madison Avenue at Punch Bowl Road
Morristown, New Jersey 07960

Subject: FUEL DENSIFICATION

Gentlemen:

Transmitted herewith are (1) an Order by the Director of Regulation changing the Technical Specifications of License No. DPR-16; and (2) copies of supporting documentation.

It is requested that by 12:00 p.m. (noon EST) August 27, 1973, you inform the Commission by Telephone and telegraph of the actions you have taken to comply with the Order for Modification of License and the maximum reactor power level that can be attained consistent with the Order.

Sincerely,

Original Signed by
A. Giambusso

A. Giambusso, Deputy Director
for Reactor Projects
Directorate of Licensing

Enclosures:
1. Order for Modification of License
2. Technical Report on Densification of General Electric Reactor Fuels
3. Safety Evaluation of the Fuel Densification Effects on the Oyster Creek Nuclear Power Station

cc w/enclosures:
see next page

OFFICE ▶	L:ORB#1	L:ORB#1	OGC	L:TR	L:OR
SURNAME ▶	SATsets:dc TVWambach	RJSchemel		JMHendrie	DJSkovhoJ
DATE ▶	8/24/73	8/24/73	8/24/73	8/24/73	8/24/73

318 (Rev. 9-53) AECM 0240 GPO 643—16—81465-1 445-678

cc: George F. Trowbridge, Esquire
Shaw, Pittman, Potts, Trowbridge
& Madden
910 - 17th Street, N. W.
Washington, D. C. 20006

GPU Service Corporation
ATTN: Mr. Thomas M. Crimmins
Safety & Licensing Manager
260 Cherry Hill Road
Parsippany, New Jersey 07054

J. Lester Yoder, Jr., Esquire
206 Horner Street
Toms River, New Jersey 08753

Mr. Kenneth B. Walton
Brigantine Tutoring
309 - 21st Street, South
Brigantine, New Jersey 08203

Miss Dorothy R. Horner
Township Clerk
Township of Ocean
Waretown, New Jersey 08753

Ocean County Library
15 Hooper Avenue
Toms River, New Jersey 08753

UNITED STATES OF AMERICA
ATOMIC ENERGY COMMISSION

In the Matter of	)	
	)	
JERSEY CENTRAL POWER & LIGHT CO.	)	Docket No. 50-219
	)	
(Oyster Creek Nuclear Power Plant)	)	

ORDER FOR MODIFICATION OF LICENSE

I.

The Jersey Central Power & Light Co. ("the licensee") is the holder of Facility License DPR-16. License DPR-16 authorizes operation of the Oyster Creek Nuclear Power Station ("the plant") in Ocean County, New Jersey. This license expressly provides, inter alia, that it is subject to all rules, regulations and orders of the Commission now or hereafter in effect.

II.

On November 14, 1972, the AEC Regulatory Staff ("the Staff") issued a report entitled "Technical Report on Densification of Light Water Reactor Fuels" ("the Report"). By letter of November 20, 1972, the Staff requested the licensee to submit analyses and data specified in the report related to determining the consequences of fuel densification for normal operation of the plant, for operation of the plant during various maneuvers and transients, and under postulated accident situations, including the design basis loss-of-coolant accidents. On January 18, 1973, the licensee provided the requested information including, by reference, the General Electric Company Report NEDM-10735, "Densification Considerations in BWR Fuel Design and Performance," dated December, 1972. The Staff reviewed the licensee's submissions, as well as Facility Change Request No. 4 and its supplements No. 1, 3 and 4, and five additional supplements to NEDM-10735 which were submitted by the General Electric Company in response to requests for additional information from the Staff. The latest of these supplements was dated July, 1973. By letter of July 16, 1973, the Staff requested the licensee, inter alia, to furnish additional analyses regarding the calculated peak cladding temperatures during a postulated loss-of-coolant accident. On August 15, 1973, the licensee submitted the requested information including Supplement 6 to NEDM-10735 and "Exxon Nuclear Oyster Creek Densification Analysis."

On the basis of the Staff's review of the above identified submittals and its evaluation of fuel densification effects upon the operation of boiling water reactors which are reflected in a safety evaluation report relating to the plant dated August 24, 1973, the Staff has determined that changes in the operating conditions for the plant are necessary in order to assure that the calculated peak cladding temperature of the core of the plant following a postulated loss-of-coolant accident will not exceed 2300°F taking into account fuel densification effects as described in the Staff's safety evaluation identified above, and, therefore, that the Technical Specifications of License DPR-16 should be amended to require: (1) the immediate control of steady-state power operation so that the average linear heat generation of all the rods in any fuel assembly, as a function of planar exposure, at any axial location, shall not exceed the maximum average planar linear heat generation rate defined by the curves in Limiting Condition for Operation, figure 3.10.1, of section 3.10.A. of the attached Appendix I, attached hereto; and (2) that during steady state power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed the maximum allowable LHGR as calculated using the equation for maximum LHGR provided in Limiting Condition for Operation, section 3.10.B. of the attached Appendix I.

III.

In view of the foregoing, the Director of Regulation finds that the public health, safety, and interest require that the following Order be made effective immediately. Pursuant to the Atomic Energy Act of 1954, as amended, the Commission's regulations in 10 CFR §§ 2.204 and 50.100 and the license condition noted in Part I above.

IT IS ORDERED THAT:

The Technical Specifications of License DPR-16 are hereby changed, to include Limiting Conditions for Operation, sections 3.10.A. and 3.10.B., and Surveillance Requirements, sections 4.10.A. and 4.10.B. attached hereto as Appendix I and the plant shall be operated immediately in accordance therewith.

IV.

Within thirty (30) days from the date of publication of this notice in the Federal Register the licensee may file a request for a hearing with respect to this Order. Within the same thirty (30) day period any other person whose interest may be affected may file a request for a hearing with respect to this Order in accordance with the provisions of 10 CFR § 2.714 of the Commission's Rules of Practice. If a request for a hearing is filed within the time prescribed herein, the Commission will issue a notice of hearing or an appropriate order.

For further details pertinent to this Order see: the Staff Technical Report on Densification of Light Water Reactor Fuels, November 14, 1972; letter to R. H. Sims from A. Giambusso, November 20, 1972; letter to A. Giambusso from R. H. Sims, January 18, 1973, with enclosure General Electric topical report, Densification Considerations in BWR Fuel Design and Performance; letter to R. H. Sims from R. J. Schemel, with enclosure the Staff's GE Model for Fuel Densification, July 16, 1973; letter to R. J. Schemel from I.R. Finfrock, Jr., August 15, 1973; the Staff Technical Report of Densification of General Electric Reactor Fuels, August 23, 1973; the Staff Technical Report of Densification of Exxon Nuclear Company Reactor Fuels (to be issued by September 4, 1973), the Staff Safety Evaluation of the Fuel Densification Effects on the Oyster Creek Nuclear Power Station, August 24, 1973; all of which are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C.

Copies of these documents may be obtained upon request addressed to the Deputy Director for Reactor Projects, Directorate of Licensing, U. S. Atomic Energy Commission, Washington, D. C. 20545.

FOR THE ATOMIC ENERGY COMMISSION

Director of Regulation

Dated at Bethesda, Maryland
this 24th day of August, 1973

APPENDIX I TO AEC ORDER

CHANGE NO. 16 TO THE TECHNICAL SPECIFICATIONS

LICENSE NO. DPR-16

JERSEY CENTRAL POWER AND LIGHT COMPANY

DOCKET NO. 50-219

AUGUST 24, 1973

3.10 Power Distribution

Applicability:

Applies to limiting the local linear heat generation rate and the average planar linear heat generation rate.

Objective:

To assure conformance to the peak clad temperature limitations during a postulated loss-of-coolant accident as specified in the Interim Acceptance Criteria and to assure conformance to the 17.2 kW/ft design limit for local linear heat generation rate.

Specification:

A. Average Planar LHGR

During steady state power operation, the average linear heat generation rate (LHGR) of all the rods in any fuel assembly, as a function of average planar exposure, at any axial location shall not exceed the maximum average planar LHGR shown in Figure 3.10.1.

B. Local LHGR

During steady state power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly, at any axial location shall not exceed the maximum allowable LHGR as calculated by the following equation:

$$LHGR \leq LHGR_d \left[1 - \left(\frac{\Delta P}{P}\right)_{max} \cdot \left(\frac{L}{LT}\right)\right]$$

Where: $LHGR_d$ = Design LHGR = 17.2 kW/ft

$\frac{\Delta P}{P}$ = Maximum Power Spiking Penalty

= 0.038 for Type I fuel
= 0.032 for Type II fuel
= 0.046 for Type III fuel
= 0.046 for Type III E fuel

LT = Total core length = 144 inches

L = Axial position above bottom of core

Basis:

The specification for average planar LHGR assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the 2300°F limit specified in the Interim Acceptance Criteria (IAC) issued in June 1971 considering the postulated effects of fuel pellet densification.

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of fuel assembly at any axial location and is only dependent secondarily on the rod to rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than ±20°F relative to the peak temperature for a typical fuel design, the limit on the average linear heat generation rate is sufficient to assure that calculated temperatures are below the IAC limit.

The maximum average planar LHGR shown in Figure 3.10.1 for Type I and Type II fuel are the same as that shown on the curves labeled "γ"(Gamma) on Figure 4-9A1 and 4-9A2 of the GE topical report "Fuel Densification Effects on General Electric Boiling Water Reactor Fuel," NEDM-10735, Supplement 6, August 1973 and are the result of the calculations presented in Section 4.3.4 of the same report. These calculations were made to determine the effect of densification on peak clad temperature and were performed in accordance with the AEC Fuel Densification Model for BWRs which is attached to NEDM-10735, Supplement 6 as Appendix B.

The maximum average planar LHGR shown in Figure 3.10.1 for Type III and Type III E fuel represents the result of the staff's independent analysis of the response of Exxon Nuclear Company's fuels under loss-of-coolant accident conditions. For this analysis, the staff has applied a 100°F allowance to account for difference in geometry between the Exxon fuel assembly and the fuel assembly for which spray cooling heat transfer data were obtained.

The possible effects of fuel pellet densification were: (1) creep collapse of the cladding due to axial gap formation; (2) increase in the LHGR because of pellet column shortening; (3) power spikes due to axial gap formation; and (4) changes in stored energy due increased radial gap size. Calculations show that clad collapse is conservatively predicted not to occur during the current power operation cycle (Cycle 3). Therefore, clad collapse is not considered in the analyses. Since axial thermal expansion of the fuel pellets is greater than axial shrinkage due to densification, the analyses of peak clad temperature do not consider any change in LHGR due to pellet column shortening. Although, the formation of axial gaps might produce a local power spike at one location on any one rod in a fuel assembly, the increase in local power density would be on the order of only 2% at the axial midplane. Since small local variations in power distribution have a small effect on peak clad temperature, power spikes were not considered in the analysis of loss-of-coolant accidents. Changes in gap size affect the peak clad temperature by their effect on pellet clad thermal conductance and fuel pellet stored energy. The pellet-clad thermal conductance assumed for each rod is dependent on the steady state operating linear heat generation rate and the gap size. As specified in the AEC Fuel Densification Model for BWR's, the gap size was calculated assuming that the pellet densified from the measured pellet density to 96.5% of theoretical density.

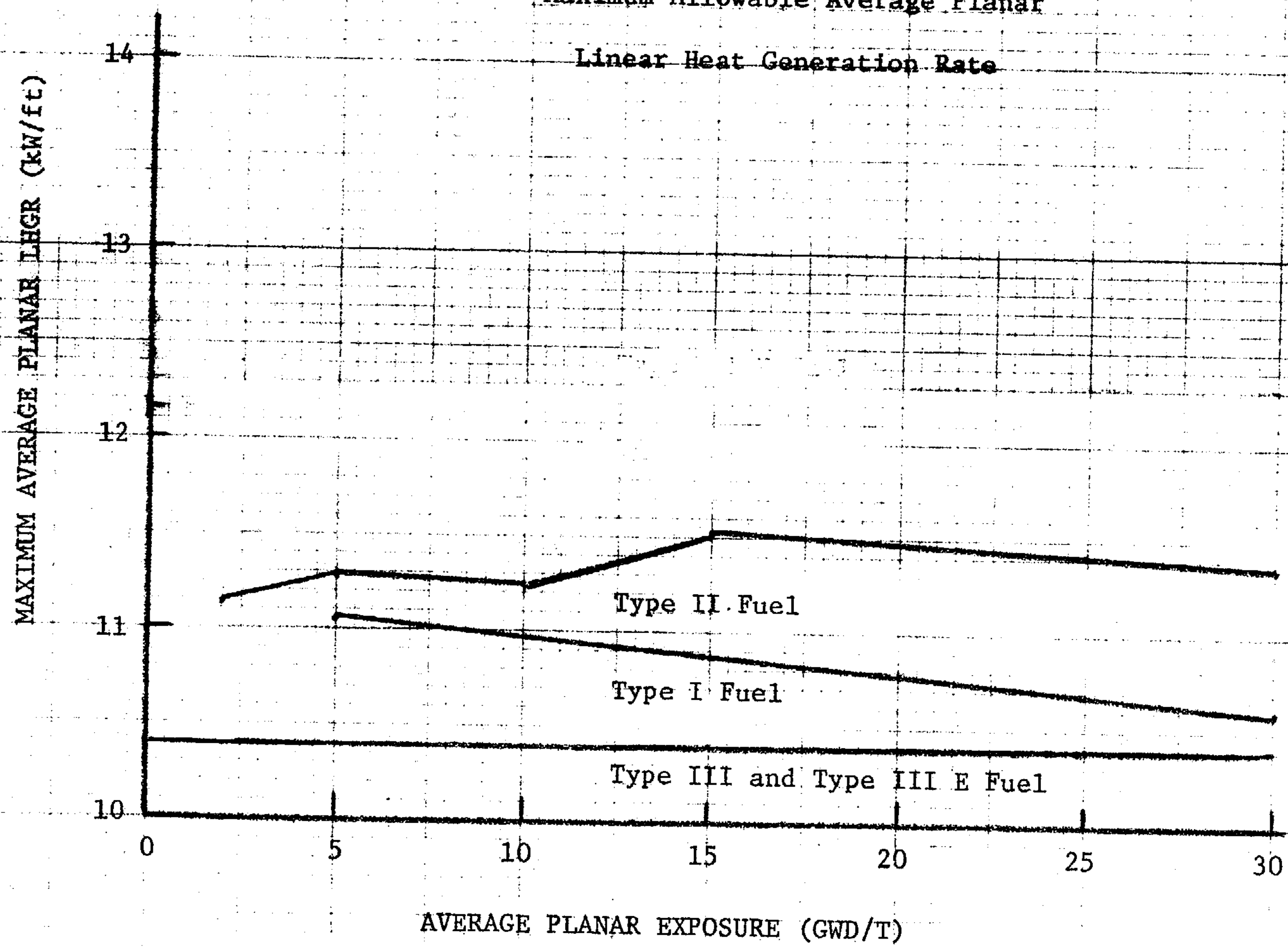
The curves used to determine pellet-clad thermal conductance as a function of linear heat generation are based on experimental data and predict with a 95% confidence that 90% of the population exceed the predictions.

This specification for local LHGR assures that the linear heat generation rate in any rod is less than the design linear heat generation even if fuel pellet densification is postulated. The power spike penalty specified for Type I and II fuel is based on the analysis presented in Section 3.2.1 of the GE topical report NEDM-10735 Supplement 6 and in Section I.A of Attachment 1 to reference 11 for Type III and III E fuel, and assumes a linearly increasing variation in axial gaps between core bottom and top, and assures with 95% confidence that no more than one fuel rod exceeds the design linear heat generation rate due to power spiking.

Figure 3.10.1

Maximum Allowable Average Planar

Linear Heat Generation Rate



4.10 Power Distribution

Applicability:

Applies to the periodic measurement of the power distribution in the core during power operation.

Objective:

To assure that the limits of Section 3.10 are not being violated.

Specification:

A. Average Planar LHGR

Daily during reactor power operation, the average planar LHGR shall be checked.

B. Local LHGR

Daily during reactor power operation, the local LHGR shall be checked.

Basis:

The LHGR shall be checked daily to determine whether fuel burnup, or control rod movement has caused changes in power distribution. Since changes due to burnup are slow, and only a few control rods are moved daily, a daily check of power distribution is adequate.

SAFETY EVALUATION OF THE
FUEL DENSIFICATION EFFECTS ON THE
OYSTER CREEK NUCLEAR GENERATING STATION

DOCKET NO. 50-219

AUGUST 24, 1973

Regulatory Staff

U. S. ATOMIC ENERGY COMMISSION

INTRODUCTION

Since the issuance of Provisional Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station on April 9, 1969, the phenomenon of fuel pellet densification has been observed in operating reactors. Based on the information initially available, the staff issued a report on November 14, 1972, entitled "Technical Report on Densification of Light-Water Reactor Fuels" (Ref 1). In this report the staff concluded that the effect that densification might have on normal operation, transients, and accidents should be evaluated for all water-cooled nuclear power plants. This conclusion was implemented by letters to the licensee on November 20, 1972 and July 16, 1973, that requested the licensee to provide the necessary analyses and other relevant information to determine the consequences of densification and its effect on normal operation, transients and accidents.

On January 17, 1973, General Electric (GE) submitted the topical report "Densification Considerations in BWR Fuel Design and Performance," NEDM-10735 (Ref 2) which provided the requested information as it applied to GE boiling water reactors generally. Subsequently, GE submitted five supplements (Ref 3, 4, 5, 6 and 7) to this topical report which provided additional information. Based on this information the Regulatory staff issued the report entitled "Technical Report on Densification of General Electric Reactor Fuels" (Ref 8). The Regulatory staff will issue by September 4, 1973, a review of the Exxon Nuclear fuel entitled "Technical Report on Densification of Exxon Nuclear Company Reactor Fuels." The licensee provided analyses of the effect of densification on steady

INTRODUCTION

Since the issuance of Provisional Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station on April 9, 1969, the phenomenon of fuel pellet densification has been observed in operating reactors. Based on the information initially available, the staff issued a report on November 14, 1972, entitled "Technical Report on Densification of Light-Water Reactor Fuels" (Ref 1). In this report the staff concluded that the effect that densification might have on normal operation, transients, and accidents should be evaluated for all water-cooled nuclear power plants. This conclusion was implemented by letters to the licensee on November 20, 1972 and July 16, 1973, that requested the licensee to provide the necessary analyses and other relevant information to determine the consequences of densification and its effect on normal operation, transients and accidents.

On January 17, 1973, General Electric (GE) submitted the topical report "Densification Considerations in BWR Fuel Design and Performance," NEDM-10735 (Ref 2) which provided the requested information as it applied to GE boiling water reactors generally. Subsequently, GE submitted five supplements (Ref 3, 4, 5, 6 and 7) to this topical report which provided additional information. Based on this information the Regulatory staff issued the report entitled "Technical Report on Densification of General Electric Reactor Fuels" (Ref 8). The Regulatory staff will issue by September 4, 1973, a review of the Exxon Nuclear fuel entitled "Technical Report on Densification of Exxon Nuclear Company Reactor Fuels." The licensee provided analyses of the effect of densification on steady state operations, operating transients and postulated accidents at the Oyster Creek Nuclear Generating Station in their letter of August 15, 1973 and the referenced GE topical report "Fuel Densification Effects on General Electric Boiling Water Reactor Fuel," NEDM-10735, Supplement 6, August 1973 (Ref 9). A discussion of our review of fuel densification as it applies to the OCNGS and our evaluation of the analyses of steady state operation, operating transients and postulated accidents is presented in subsequent sections of this report.

DENSIFICATION EFFECTS

A detailed discussion of the causes and effects of densification including the results of observations of irradiated fuel in both test and power reactor fuel, an investigation of the possible mechanisms and evaluation of the controlling parameters, is presented in the staff reports on densification (Ref 1, 8 and 12). At this time the only clear conclusion that can be drawn is that under irradiation fuel pellets can shrink and decrease in volume with corresponding changes in pellet dimensions. Four principal effects are associated with the dimensional changes resulting from densification. A decrease in length of pellets could result in the formation of axial gaps in the column of fuel pellets within a fuel rod. Two effects are associated with axial gaps. First, if relatively large axial gaps form, creepdown of the cladding later in life may lead to collapse of the cladding into the gaps. Second, axial gaps produce a local increase in the neutron flux and generate a local power spike. A third effect, which results from a decrease in pellet length, is a directly proportional increase in linear heat generation rate.

A decrease in pellet radius could result in the increase in the radial clearance between the fuel pellet and the fuel rod cladding. A fourth effect, which results from a decrease in pellet radius, is decreased pellet-clad thermal conductance (gap conductance). Decreased conductance would increase the fuel pellet temperature and stored energy and decrease the heat transfer capability of the fuel rod. Each of these four effects has been considered in evaluating the total effect that fuel densification might have on normal operation, transients and accidents.

Based on experimental evidence that no collapse has been observed in BWR fuel rods and on the results of calculations performed independently by the staff and GE, the Regulatory staff has concluded that typical BWR fuel will not collapse during the first cycle of operation (Section 3.4.2, Ref 8). GE and Exxon Nuclear have also calculated the creep collapse of fuel in later cycles using a model which includes the modifications specified by the staff (Section 3.4.2, Ref 8). The results of these calculations for fuel in residence up to September 1974 are reported in Supplement 6 of the GE report (Ref 9) and in reference 10 for Exxon Nuclear fuel and indicate that clad collapse will not occur. The staff has reviewed the GE calculations and performed independent calculations, which also predict that collapse will not occur. Based on the calculations and experimental evidence, the staff concludes that creep-collapse need not be considered as affecting normal operation, transients or accidents.

The increase in linear heat generation rate (LHGR) resulting from contraction of the fuel is offset by compensating factors. Although pellets with initial densities less than the mean initial density will contract more than the average pellet, such pellets also contain correspondingly less fuel and produce less power in a given neutron flux. Therefore, only contraction from an initial mean pellet density need be considered in determining the LHGR. This contraction is offset by thermal expansion, as shown by calculations summarized in Table 3-1 of Supplement 6 of the GE report (Ref 9) and in Section I B of Attachment 1 to reference 11. Since the increase in fuel column length due to thermal expansion was not considered in the original design calculations or transient and accident analyses, and since the effect of thermal expansion offsets the effect of densification on LHGR, it is appropriate to use the design LHGR in the analyses of normal operation, transients and accidents when considering the effects of densification. This was done in all the analyses presented by GE in Supplement 6 of the topical report (Ref 9) and Exxon in reference 11.

Calculations by GE and Exxon of power spikes resulting from possible axial gaps in the fuel take into account the peaking due to a given gap, the probability distribution of peaking due to the distribution of gaps, and the convolution of the peaking probability with the design radial power distribution. Based on an examination of the methods used, comparison with requirements and approved models given in the staff densification report, and check calculations performed for the staff by Brookhaven National Laboratory, the staff concluded in their report (Ref 8) that, if appropriate gap assumptions are made regarding sizes, the GE calculational method is acceptable. The staff also concluded in their report (Ref 12) that the Exxon calculational method is acceptable if appropriate gap distribution and assembly peaking assumptions are made. The results of calculations of power spikes using acceptable gap sizes are summarized in Figure 3-6 of Supplement 6 of the GE report (Ref 9) and in Section I of reference 11 for Exxon fuel. During normal operation there is a 95% confidence that no more than one rod would have a power spike greater than approximately 4% at the top of the fuel. At the midplane the corresponding power spike would be approximately 2%. When the reactor power is low and there are no voids, the spike could be greater. Under these conditions, there is a 95% confidence that no more than one rod would have a power spike greater than 5% at the top of the fuel.

Pellet-clad thermal conductance is a function of gap size and linear heat generation rate. The staff has reviewed the experimental data and analyses that GE has submitted to justify their correlation of gap conductance, examined the uncertainties in the data, and performed independent calculations with a fuel thermal performance computer program. The pellet-clad thermal conductance correlation used by GE is depicted in Figure 3-10 of Supplement 6 of the GE report (Ref 9) and that used by Exxon is presented in Section III c of reference 11. They are based on experimental data and predicts with a 95% confidence that 90% of the total population of pellet-clad conductances exceed the prediction. The staff concludes that these correlations when used with a gap size adjusted for the effects of densification are acceptable.

EVALUATION OF EFFECTS OF DENSIFICATION

Normal Operation

The design limits affected by fuel densification are the design values of linear heat generation rate (LHGR) and minimum critical heat flux ratio (MCHFR). The power spike resulting from axial gaps is considered in limiting operation of the reactor. The Technical Specifications will require that the LHGR in any rod at any axial location be less than the design value of 17.2 kw/ft by a margin equal to or greater than the power spike calculated using the acceptable model. As discussed previously, this power spike penalty will assure at the 95% confidence level that no more than one rod will exceed the design value LHGR. Since the random occurrence of local power spikes will have no effect on coolant flow or quality, the uncertainty in calculation of the critical heat flux is unchanged. Therefore, if the calculated MCHFR is maintained above the steady state design limit of 1.9 and the margin to the design value of LHGR is also maintained, the probability of reaching a MCHFR of 1.0 is essentially unchanged from that calculated in the FSAR.

Transient Performance

The key transients for evaluation of BWR performance are those associated with overpressurization, which might imperil the integrity of the primary coolant pressure boundary, and with reduction of coolant flow, which might imperil the integrity of the fuel clad. The transient resulting from a turbine trip without opening the bypass valves is representative of transients that might result in overpressurization. The transient resulting from the simultaneous trip of the recirculation pump drive motors is representative of transients that result in a rapid reduction of core flow.

Following isolation of a BWR, such as would result from closure of the turbine stop and bypass valves, stored and decay energy from the core increases the coolant temperature and pressure. Since densification might reduce the pellet-clad conductance and increase the stored energy, densification could effect the peak pressure following a transient. GE has calculated the increase in heat flux, fuel temperature and peak pressure in the primary coolant system following a turbine trip transient without bypass using gap conductances as low as 400 Btu/hr-ft^2-°F (Ref 9). A conductance of 400 Btu/hr-ft^2-°F is representative of the average fuel rod and its use is appropriate since the average fuel rod stored energy is the appropriate parameter to use when evaluating coolant system pressure. The calculated peak pressure is increased only 5 psi and is not significantly greater than the system pressure calculated using the value of 1000 Btu/hr-ft^2-°F for gap conductance. Using a conductance of 400 Btu/hr-ft^2-°F increased the calculated fuel temperature 13°F and the heat flux 1%. These increases are also insignificant.

Following a rapid reduction in core flow, such as would result from simultaneously tripping the recirculation pump motors, the MCHFR will decrease. A MCHFR of 1.0 is taken as a design limit for fuel damage. The slower thermal response of rods with densified fuel can result in a lower MCHFR following a rapid flow reduction. GE has calculated that the heat flux at the time of MCHFR would increase less than 5%, even if the gap conductance were as low as 400 Btu/hr-ft^2-°F. This conductance is representative of the lower bound of the conductance expected at the axial location where MCHFR occurs.

Based on these calculations, the staff concludes that changes in gap conductance resulting from fuel densification would affect the course of flow and pressure transients. However, the pressure or MCHFR limits would not be exceeded.

Refueling Accident

Since fuel densification does not affect any parameters used in the evaluation of the refueling accident, the consequences of this accident are unchanged.

Control Rod Drop Accident

A generic evaluation by the staff of the control rod drop accident has been underway for the past several months. General Electric has submitted topical reports revising the techniques for analyses of the control rod drop accident including, among other features, a change in the method for modeling the rate of negative reactivity insertion. These topical reports and revised analyses are under review. However, the parameters important to the analysis such as gross power distribution, delayed neutron fraction and the reactivity changes produced by the dropped rod, the scram insertion of the other rods and Doppler feedback are not significantly affected by densification. The parameters affected by densification are initial stored energy and heat transfer. These factors are not important for the control rod drop accident at low reactor power which results in the largest energy deposition, since the analysis assumes low power and adiabatic fuel pins and therefore, no stored energy and no heat transfer. From our independent calculations we have concluded that the transient effects of a rod drop accident while operating at power levels above 20% would be also small.

Main Steam Line Break Accidents

As in the analysis of transients, the effect of reduced gap conductance resulting from densification is an increase in stored energy and transient heat flux. However, calculations demonstrate that a reduced conductance does not result in departure from nucleate boiling during the transient (Ref 9). As in the calculation presented in the FSAR (gap conductance equal 1000 Btu/hr-ft^2-°F) no clad heatup is predicted to occur and consequently, the main steam line break accident is unaffected by densification.

Loss of Coolant Accident

Small Break

As in the analysis of a transient, the effect of reduced gap conductance resulting from densification is an increase in stored energy and transient heat flux. A higher initial stored energy, when transferred to the coolant during blowdown, maintains the pressure, and increases the break flow rate resulting in a quicker actuation of the Automatic Depressurization System. Therefore, the reactor is depressurized sooner and the low pressure emergency core cooling systems refill the vessel sooner. Since all stored energy is removed during the initial phase of the blowdown, only the decay heat, which is the same in both cases, affects the clad temperature. The net effect is a reduction in peak clad temperature following a small pipe break. Therefore, densification does not adversely affect a small pipe break accident.

Design Basis LOCA

Following a postulated break of a recirculation pipe, densification can affect the hydraulic response of the reactor as calculated by the blowdown analysis and the thermal response of the fuel as calculated by the heatup model. The effect on the blowdown is much less significant than the effect during the heatup.

The staff has not completed its review of the response of the Exxon Type III and Type III E fuel bundles under the design basis loss-of-coolant accident transient. The staff has concluded, however, that a conservative approach to evaluating the Exxon reload fuels for LOCA conditions is obtained by employing a constant limit on the maximum average planar LHGR of 10.4 kw/ft. The staff has determined this limit to be conservative based on independent staff calculations performed for the Oyster Creek plant and using the 95/90 limit curves presented in Figure 3-10 of reference 9. For this analysis, the staff has applied a 100°F allowance to account for difference in geometry between the Exxon fuel assembly and the fuel assembly for which spray cooling heat transfer data were obtained.

As discussed in the review of the transient analysis, the effect of densification is a reduction of gap conductance and a corresponding increase in stored energy and transient heat flux. The increased energy and heat flux result in a slightly modified hydraulic response following the LOCA. However, as shown in Figures 4-7 and 4-8 of Supplement 6 to the GE report (Ref 9), the flow rates are not significantly changed and the time of departure from nucleate boiling is unchanged. Therefore, the convective heat transfer coefficients are not significantly changed as a result of densification.

The heatup of the fuel is, however, significantly changed primarily as a result of increased stored energy. Although the formation of axial gaps might produce a local power spike, as discussed previously the spike would be approximately 2% at the axial midplane. As discussed in the staff report (Section 4.3, Ref 8), it is improbable that more than one spike of significant magnitude would occur at any axial elevation and that a 1% power spike would result in only a 4°F increase in peak clad temperature. Therefore, the effect of power spikes can be neglected in the heatup analysis.

The peak clad temperature following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate and stored energy of all the rods in a fuel assembly at the axial location corresponding to the peak of the axial power distribution. GE has calculated (p. 4-12, Ref 9) that expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than ±20°F relative to the peak temperature for a typical fuel design. Staff calculations (Table II, Ref 8) show that variations in individual gap conductances and therefore, stored energy within an assembly result in peak clad temperatures approximately 20°F higher than temperatures calculated using only the conductance of the average rod to represent all the rods.

The stored energy is dependent on the LHGR and the pellet-clad thermal conductance. As discussed, the conductance is based on a correlation which underpredicts 90% of the data with a 95% confidence for a selected gap size. The gap size is calculated as specified in the AEC Fuel Densification Model assuming that the pellet densified from the initial density to 96.5% of theoretical density. Since peak clad temperature is primarily a function of average stored energy, the density of 48 rods is taken as the two standard deviation lower bound on the measured initial "boat" pellet density. For the most critical rod, the two standard deviation lower bound on initial density of individual pellets was assumed. The result of calculations of peak clad temperature are presented in Figure 4-10A of Supplement 6 to the GE report (Ref 9). The staff concludes that limitation of the average linear heat generation rate of all the rods in any GE fuel assembly at any axial location to the values of the curves labeled "γ" in Figures 4-9A1 and 4-9A2 of reference 9, and the limitation of the average linear heat generation rate of all rods in any Exxon fuel assembly at any axial location to a value of 10.4 kW/ft, will assure that calculated peak clad temperatrues will not exceed 2300°F.

CONCLUSIONS

The Regulatory staff has reviewed the General Electric Co. report "Fuel Densification Effects on General Electric Boiling Water Reactor Fuel," NEDM-10735 (Ref 2, 3, 4, 5, 6, 7, 9) for its applicability to the Oyster Creek Nuclear Generating Station. The staff concludes that the following changes in the operating conditions for Oyster Creek Nuclear Generating Station are necessary in order to assure that the calculated peak cladding temperature of the core following a postulated LOCA will not exceed 2300°F taking into account fuel densification effects: (1) the immediate control of steady-state power operation so that the average linear heat generation of all the rods in any fuel assembly, as a function of planar exposure, at any axial location, shall not exceed the maximum average planar linear heat heat generation rate defined by the curvesin Limiting Condition for Operation, Figure 3.10.1, of Section 3.10.A of the Appendix I, attached to Order for Modification of License, dated August 24, 1973, and (2) that during steady state power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed the maximum allowable LHGR as calculated using the equation for maximum LHGR provided in Limiting Condition for Operation, Section 3.10.A of Appendix I attached to the Order.

References

1. "Technical Report on Densification of Light-Water Reactor Fuels," by the USAEC Regulatory staff, November 14, 1972.

2. "Densification Considerations in BWR Fuel Design and Performance" NEDM-10735, December 1972.

3. "Response to AEC Questions - NEDM-10735," NEDM-10735, Supplement 1, April 1973.

4. "Responses to AEC Questions NEDM-10735 Supplement 1," NEDM-10735 Supplement 2, May 1973.

5. "Responses to AEC Questions NEDM-10735 Supplement 1," NEDM-10735 Supplement 3, June 1973.

6. "Responses to AEC Questions NEDM-10735" NEDM-10735 Supplement 4, July 1973.

7. "Densification Considerations in BWR Fuel" NEDM-10735, Supplement 5, July 1973.

8. "Technical Report on Densification of General Electric Reactor Fuels," August 23, 1973, Regulatory staff, USAEC.

9. "Fuel Densification Effects on General Electric Boiling Water Reactor Fuel," NEDM-10735, Supplement 6, August 1973, General Electric Co.

10. "Cladding Collapse Calculational Procedure" JN-72-23 November 1972, K.R. Merckx.

11. Response to AEC Questions. Jersey Central letter dated August 15, 1973.

12. "Technical Report on Densification of Exxon Nuclear Company Reactor Fuels" to be issued by September 4, 1973, Regulatory staff, USAEC.