Distribution:
Docket File
AEC PDR
DR Reading
L Reading
RP Reading
ORB-2 File
JBuchanan, ORNL
TAbernathy, DTIE
EPA (3)
LMuntzing RReid
JO'Leary MJinks (4)
AGiambusso NDube
JHendrie SKari
VStello RDiggs
DSkovholt WMiller
ACRS (16)
RO (3)
OGC
DZiemann
RVollmer
TCarter
JRiesland
SVarga

AUG 24 1973

Docket Nos. 50-254
and 50-265

Commonwealth Edison Company
ATTN: Mr. J. S. Abel
Nuclear Licensing Administrator -
Boiling Water Reactors
Post Office 767
Chicago, Illinois 60690

Subject: FUEL DENSIFICATION

Gentlemen:

Transmitted herewith are (1) an Order by the Director of Regulation changing the Technical Specifications of License Nos. DPR 29 and DPR-30; 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 Quad-Cities Units 1 and 2

cc w/enclosures: (see next page)

OFFICE ▶	L:ORB-2	L:ORB-2	OGC	L:PR/DD	L:OR/AD	L:RP/DD
SURNAME ▶	RDiggs:vvw JRiesland	DZiemann		JHendrie	DSkovholt	AGiambusso
DATE ▶	8/24/73	8/24/73	8/24/73	8/24/73	8/24/73	8/ /73

cc: John W. Rowe, Esquire
Isham, Lincoln & Beale

Moline Public Library

Mr. Charles Whitmore
President and Chairman
Iowa-Illinois Gas and
Electric Company

OFFICE ►						
SURNAME ►						
DATE ►						

Distribution

Docket File	DJSkovholt
SEC PDR	ACRS (16)
DR Reading	RO (3)
L Reading	OGC
RP Reading	DLZiemann
Branch Reading	RHVollmer
JRBuchanan, ORNL	TJCarter
TBAbernathy, DTIE	JIRiesland
EPA (3)	SVarga
LMMuntzing	RReid
JFO'Leary	MJinks (4)
AGiambusso	NDube
JMHendrie	SKari
VStello	RMDiggs
	WOMiller

Docket Nos. 50-254
and 50-265

Commonwealth Edison Company
ATTN: Mr. J. S. Abel
Nuclear Licensing Administrator -
Boiling Water Reactors
Post Office 767
Chicago, Illinois 60690

Subject: FUEL DENSIFICATION

Gentlemen:

Transmitted herewith are (1) an Order by the Director of Regulation changing the Technical Specifications of License Nos. DPR-29 and DPR-30; 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 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,

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 Quad-Cities Units 1 and 2

cc w/enclosures:
See page 2

OFFICE ▸	L:ORB#2	L:ORB#2	OGC	L:TR	L:OR	L:RP
SURNAME ▸	RMDiggs:cls JIRiesland	DLZiemann		JMHendrie	DJSkovholt	AGiambuss
DATE ▸	8/23/73	8/ /73	8/ /73	8/ /73	8/ /73	8/ /73

Form AEC-318 (Rev. 9-53) AECM 0240

UNITED STATES OF AMERICA
ATOMIC ENERGY COMMISSION

In the Matter of	)	
	)	
COMMONWEALTH EDISON CO.	)	Docket Nos. 50-254
	)	50-265
(Quad-Cities Nuclear Power Station,	)	
Units 1 and 2)	)	

ORDER FOR MODIFICATION OF LICENSE

I.

The Commonwealth Edison Co. ("the licensee") is the holder of Facility Licenses DPR-29 and DPR-30. Licenses DPR-29 and DPR-30 authorize operation of the Quad-Cities Nuclear Power Station, Units 1 and 2 ("the plants") in Rock Island County, Illinois. These licenses expressly provide, inter alia, that they are subject to all rules, regulations and orders of the Commission now or hereinafter 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 plants, for operation of the plants during various maneuvers and transients,

and under postulated accident situations, including the design basis loss-of-coolant accidents. On January 3, 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 submission as well as 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.

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 plants are necessary in order to assure that the calculated peak cladding temperature of the core of the plants 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 Licenses DPR-29 and DPR-30 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 curve in Limiting Condition for Operation, figure 3.5.1, of section 3.5.J. 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 3.5.K. 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 Licenses DPR-29 and DPR-30 are hereby changed, to include Limiting Conditions for Operation, sections 3.5.J. and 3.5.K., and Surveillance Requirements, sections 4.5.J. and 4.5.K. 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 B. Lee, Jr. From A. Giambusso, November 20, 1972; letter to A. Giambusso from L. D. Butterfield, Jr., January 3, 1973, with enclosure General Electric topical report, Densification Considerations in BWR Fuel Design and Performance; letter to B. Lee, Jr., from D. Zieman, with enclosure the Staff's

GE Model for Fuel Densification, July 16, 1973; letter to D. Zieman from J. S. Abel, August 15, 1973; the Staff Technical Report for Densification of General Electric Reactor Fuels, August 23, 1973; the Staff Safety Evaluation of the Fuel Densification Effects on the Quad-Cities Nuclear Power Station, Units 1 and 2, 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

S. Dean Ornstein

Director of Regulation

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

SAFETY EVALUATION OF THE FUEL DENSIFICATION EFFECTS ON THE QUAD CITIES NUCLEAR POWER STATION DOCKET NOS. 50-254 AND 50-265

Regulatory Staff

U.S. Atomic Energy Commission

August 24, 1973

INTRODUCTION

Since the issuance of the Regulatory staff's Safety Evaluation for the Quad Cities Nuclear Power Station on August 25, 1971, 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 data needed 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 licensee provided analyses of the effect of densification on steady state operations, operating transients and postulated accidents at the Quad Cities Nuclear Power 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 Quads Cities Nuclear Power Station 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 obersvations 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 and 8). 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, reference 8). GE has 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, reference 8). The results of these calculations for fuel in residence up to September 1974 are reported in Supplement 6 of the GE report (reference 9) 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). 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).

Calculations by GE 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 results of calculations of power spikes using acceptable gap sizes are summarized in Figures 3.6 of Supplement 6 of the GE report (Ref. 9). 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). It is based on experimental data and preducts with a 95% confidence that 90% of the total population of pellet-clad conductances exceed the prediction. The staff concludes that this correlation when used with a gap size adjusted for the effects of densification is 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 17.5 kw/ft by a margin equal to or greater than the power spike calculated using the accepted 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 the 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 both 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 by 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 both 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 low or pressure transients. However, neither the pressure or the 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 operation at power levels above 20% would also be 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.

As discussed in the review 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, reference 8), it is improbable that more than on 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 $\pm$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 Figures 4-10K, and 4-10L 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 fuel assembly at any axial location to the values of the curve labeled "γ" in Figures 4-9K1, 4-9K2, 4-9L1, and 4-9L2 of reference 9 will assure that calculated peak clad temperatures will not exceed 2300°F.

CONCLUSIONS

The Regulatory staff has reviewed the General Electric Company 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 Quad Cities Station. The staff concludes that the following changes in the operating conditions for Quad Cities Units 1 and 2 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 generation rate defined by the curve in Limiting Condition for Operation, Figure 3.5.1, of Section 3.5.J 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 3.5.K 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, 1972, Regulatory staff, USAEC.

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