UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

July 16, 1973

Docket No. 50-219

Jersey Central Power & Light Company
ATTN: Mr. R. H. Sims
Vice President
Madison Avenue at Punch Bowl Road
Morristown, New Jersey 07960

Gentlemen:

By letter of November 20, 1972, the Commission's Regulatory staff requested that you provide the necessary analyses and other relevant data for determining the consequences of densification and its effects on normal operation, anticipated transients, and accidents, including the loss-of-coolant accident. Your response of February 22, 1973, stated that the General Electric report NEDM-10735, "Densification Considerations in BWR Fuel Design and Performance," December 1972, serves as your answer to our request.

As you are aware, five additional proprietary supplements to NEDM-10735 have been submitted by General Electric Company in response to questions raised by the staff as a result of our review of NEDM-10735 and the succeeding supplements.

By letter dated April 3, 1973, we requested additional information concerning the fuel densification analyses performed for all types of fuel in the Cycle 3 core, including that supplied by General Electric Company and that supplied by Exxon Nuclear Corporation. In response, you submitted Supplement No. 3 to Facility Change Request No. 4 dated April 17, 1973.

Enclosures A and B represent the staff's conclusions on BWR fuel densification for the GE fuel and Exxon fuel respectively and provide the essential elements to be included to account for the effects of fuel densification in the Oyster Creek core.

Therefore, we request that you provide the necessary analyses and other relevant data for determining the consequences of densification and the effects on normal operation, anticipated transients, and accidents, including the postulated loss-of-coolant accident, using the guidance provided in the enclosures. If the analyses indicate that changes in design or operating conditions are necessary to maintain required margins, you should submit proposed changes and operating limitations with the analyses.

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To permit the Regulatory staff to conduct an expeditious and orderly review of these matters, we request that you submit the analyses and additional information within thirty days from the date of this letter. It is requested that this information be provided with one signed original and thirty-nine additional copies.

Sincerely,

Robert J Schemel

Robert J. Schemel, Chief
Operating Reactors Branch No. 1
Directorate of Licensing

Enclosures:
A - Model for Fuel Densification, GE fuel
B - Model for Fuel Densification, Exxon fuel

cc: see next page

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

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

GE MODEL FOR FUEL DENSIFICATION

The General Electric fuel densification model is described in NEDM-10735 and Supplements 1, 2, 3, 4, and 5 to NEDM-10735 (see references 1 through 6). The GE model when modified as described below is considered to be suitably conservative for the evaluation of densification effects in BWR fuel.

Possible effects of fuel densification are: (1) power spikes due to axial gap formation; (2) increase in LHGR because of pellet length shortening; (3) creep collapse of the cladding due to axial gap formation; and (4) changes in stored energy due to increased radial gap size. Similarly, the GE model for fuel densification consists of four parts: power spike model, linear heat generation model, clad creep collapse model and stored energy model. The required modifications to each of these models are listed below.

Power Spike Model

The GE power spike model is acceptable as it is described in NEDM-10735 and Supplement 1 to NEDM-10735 and modified in Supplement 5 of NEDM-10735 as long as it is used in conjunction with a maximum axial gap size given by the following equation:

$$\Delta L = \left(\frac{0.965 - \rho_i}{2} + 0.004\right) L$$

where ΔL = maximum axial gap length

L = fuel column length

ρ_i = mean value of measured initial pellet density (geometric)

0.004 = allowance for irradiation induced cladding growth and axial strain caused by fuel-clad mechanical interaction

Linear Heat Generation Model

The following expression should be used to calculate the decrease in fuel column length in determinations of the linear heat generation rate:

$$\Delta L = \frac{0.965 - \rho_i}{2} L$$

where: ΔL = decrease in fuel column length

L = fuel column length

ρ_i = mean value of measured initial pellet density (geometric)

Credit can be taken for fuel column length increase due to thermal expansion, and for the actual measured length of the fuel column.

Clad Creep Collapse Model

Examination of exposed BWR fuel rods (Ref. 5) and Regulatory staff calculations show that clad collapse will not occur in typical BWR fuel during the first cycle of operation. Consequently, no additional creep collapse calculations are required for the first cycle of typical BWR fuel.

For reactors in subsequent cycles of operation the GE creep collapse model, described in NEDM-10735 and its supplements, should be used with the following modifications:

1. The equation used to calculate the change in ovality due to the increasing creep strain should account for the ovality change due to change in curvature as well as for the ovality change due to change in rod circumference.

2. A conservative value should be used for the clad temperature. Axial temperature variations in the vicinity of a fuel gap as affected by thermal radiation from the ends of the pellets and by axial heat conduction should be taken into account. Effects from any buildup of oxide and crud on the clad surfaces should also be considered.
3. The calculations should be made for the fuel rod having the worst combination of fast neutron flux and clad temperature.
4. No credit should be taken for fission gas pressure buildup.
5. No credit should be taken for end effects. An infinitely long, unsupported length of cladding should be assumed.
6. Conservative values for clad wall thickness and initial ovality should be used. An acceptable approach is to use the two standard deviation limit of as fabricated dimensions.

Stored Energy Model

The GE stored energy model is based on UO_2 thermal conductivity and heat capacity given in Section D of Reference 6, a flux depression factor of 1.0, and a gap coefficient of 1000 Btu/hr-ft^2 F applied to each fuel rod within the hot fuel assembly. The selection of the gap coefficient in this model should be modified as follows.

(1) Changes in gap conductance due to variations in LHGR, gap size (or g/D) and initial fuel pellet density should be accounted for.

(2) A gap conductance vs. LHGR curve that based on available experimental data predicts with 95 percent confidence that 90 percent of future events will exceed predictions, should be used.

(3) Instantaneous densification should be assumed, i.e., pellet OD and gap size should be calculated using the following equation:

$$\Delta r = \frac{0.965 - \rho_i + 2\sigma}{3} r$$

where: Δr = reduction in pellet radius

r = initial pellet radius

σ = standard deviation in the measured probability distribution of pellet density

ρ_i = mean value of measured initial pellet density (geometric)

The gap size and pellet, OD, corrected for instantaneous densification, should be used for the selection of the gap conductance vs. LHGR curve.

(4) The fuel pellet located at the most critical position for normal operation, anticipated transients and postulated accident conditions should be analyzed with the densified pellet size as given by the equation under item (3).

(5) In calculations which are sensitive to bundle stored energy, for the 48 neighboring pellets in the same horizontal plane, the standard deviation used in the equation can be replaced by the standard deviation in mean boat pellet density.

(6) Since the assembly average stored energy is one of the most important inputs to BWR LOCA evaluation, a Tec ical Specification limit should be imposed on maximum permitted assembly power.

References

1. D. C. Ditmore and R. B. Elkins: "Densification Considerations in BWR Fuel Design and Performance" NEDM-10735, December 1972.

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

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

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

5. "Responses to AEC Question NEDM-10735" NEDM-10735 Supplement 4, July 1973.

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

7. B. C. Slifer and J. E. Hench, "Loss-of-Coolant Accident & Emergency Core Cooling Models for General Electric Boiling Water Reactors," NEDO-10329, April 1971.

EXXON NUCLEAR MODEL FOR FUEL DENSIFICATION

The Exxon Nuclear fuel densification model is described in Reference 1. The Exxon model when modified as described below is considered to be suitably conservative for the evaluation of densification effects in BWR fuel.

Possible effects of fuel densification are: (1) power spikes due to axial gap formation; (2) increase in LHGR because of pellet length shortening; (3) creep collapse of the cladding due to axial gap formation; and (4) changes in stored energy due to increased radial gap size. Similarly, the Exxon model for fuel densification consists of four parts: power spike model, linear heat generation model, clad creep collapse model and stored energy model. The required modifications to each of these models are listed below.

Power Spike Model

The Exxon power spike model is acceptable as it is described in Reference 1 as long as it is used in conjunction with a maximum gap size given by the following equation:

$$\Delta L = \left(\frac{0.965 - \rho_i}{2} + 0.004\right) L$$

where ΔL = maximum axial gap length

L = fuel column length

ρ_i = mean value of measured initial pellet density (geometric)

0.004 = allowance for irradiation induced cladding growth and axial strain caused by fuel-clad mechanical interaction

Linear Heat Generation Model

The following expression should be used to calculate the decrease in fuel column length in determinations of the linear heat generation rate.

$$\Delta L = \frac{0.965 - \rho_i}{2} L$$

where: ΔL = decrease in fuel column

L = fuel column length

ρ_i = mean value of measured initial pellet density (geometric)

Credit can be taken for fuel column length increase due to thermal expansion, and for the actual measured fuel column length.

Clad Creep Collapse Model

Examination of exposed BWR fuel rods and Regulatory staff calculations show that the clad collapse will not occur in typical BWR fuel during the first cycle of operation. Consequently, no additional creep collapse calculations are required for the first cycle of typical BWR fuel.

For reactors in subsequent cycles of operation the Exxon creep collapse model, described in Reference 2 should be used with the following assumptions:

1. A conservative value should be used for the clad temperature. Axial temperature variations in the vicinity of a fuel gap as affected by thermal radiation from the ends of the pellets and by axial heat conduction should be taken into account. Effects from any buildup of oxide and crud on the clad surfaces should also be considered.

2. The calculations should be made for the fuel rod having the worst combination of fast neturon flux and clad temperature.
3. No credit should be taken for fission gas pressure buildup.
4. No credit should be taken for end effects. An infinitely long, unsupported length of cladding should be assumed.
5. Conservative values for clad wall thickness and initial ovality should be used. An acceptable approach is to use the two standard deviation limit of as fabricated dimensions.

Stored Energy Model

The Exxon stored energy model is based on UO_2 thermal conductivity given in Reference 1, UO_2 heat capacity given in Reference 3, a flux depression factor of 1.0, and a gap coefficient of 1000 Btu/hr-ft^2 F applied to each fuel rod within the hot fuel assembly. The selection of the gap coefficient in this model should be modified as follows:

(1) Changes in gap conductance due to variations in LHGR, gap size (or g/D) and initial fuel pellet density should be accounted for.

(2) A gap conductance vs. LHGR curve that based on available experimental data predicts with 95 percent confidence that 90 percent of future events will exceed predictions, should be used.

(3) Instantaneous densification should be assumed, i.e., pellet OD and gap size should be calculated using the following equation:

$$\Delta r = \frac{0.965 - \rho_i + 2\sigma}{3} r$$

where: Δr = reduction in pellet radius

r = initial pellet radius

ρ_i = standard deviation in the measured probability distribution of pellet density

σ = mean value of measured initial pellet density (geometric)

The gap size and pellet, OD, corrected for instantaneous densification, should be used for the selection of the gap conductance vs. LHGR curve.

(4) The fuel pellet located at the most critical position for normal operation, anticipated transients and postulated accident conditions should be analyzed with the densified pellet size as given by the equation under item (3).

(5) In calculations which are sensitive to bundle stored energy, the initial density of the 48 neighboring pellets in the same horizontal plane, should be equal to the lowest mean value of the individual pellet lot densities. To calculate the densified values, the equation under item (3) can be used substituting the lowest mean pellet lot density for ρ_i and setting the 2σ value equal to zero.

(6) Since the assembly average stored energy is one of the most important inputs to BWR LOCA evaluation, a Technical Specification limit should be imposed on maximum permitted assembly power.

References

1. Oyster Creek Nuclear Generating Station, Docket No. 50-219, Supplement No. 3 to Facility Change Request No. 4, April, 1973.

2. Merckx, K. R.: "Cladding Collapse Calculational Procedure," JN-72-23, November 1, 1972.

3. Braxfield, H. C., et. al: "Recommended Property and Reaction Kinetics Data for Use in Evaluating a Light-Water-Cooled Reactor Loss-of-Coolant Incident Involving Zircaloy-4 or 304-ss-Clad UO_2" GEMP-482, April, 1968.

J. P. O'Reilly

In reply refer to:
RO:RPB
50-219

JUL 16 1973

Jersey Central Power and Light Company
ATTN: Mr. Donald A. Ross
Manager, Nuclear Generating Stations
Madison Avenue at Punch Bowl Road
Morristown, New Jersey 07960

Gentlemen:

This will acknowledge receipt of your letter dated June 5, 1973, reporting the exposure of an individual to radioactive material. This matter will be examined during a future inspection.

Very truly yours,

Original signed by
H.D. Thornburg

John G. Davis, Deputy Director
for Field Operations
Directorate of Regulatory Operations

bcc: w/cpy ltr dtd 6/5/73
D. J. Skovholt
J. G. Keppler
J. P. O'Reilly
H. D. Thornburg
C. F. Eason
PDR
Local PDR
NSIC
DTIE
DR Reading
DR Central Files
Incident Files
RO Files

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OFFICE ▶	RO:RPB	RO	RO			
SURNAME ▶	TWBrockett:ef	GWRoy	JGDavis			
DATE ▶	7/11/73					

Jersey Central Power & Light Company

MADISON AVENUE AT PUNCH BOWL ROAD • MORRISTOWN, N. J. 07960 • 539-6111

June 5, 1973

Mr. Frank E. Kruesi, Director
Directorate of Regulatory Operations
United States Atomic Energy Commission
Washington, D. C. 20545

Dear Mr. Kruesi:

Subject: Oyster Creek Station
Docket No. 50-219
Personnel Exposure

The purpose of this letter is to advise you that during the performance of control rod drive modification and replacement, an individual, under the employ of an outside contractor, received a whole body exposure in excess of 3.0 rems. This exposure is in excess of the applicable limits as set forth in 10CAR20.101.B.1 and, as such, is being reported per 10CFR20.405.

The individual of concern was assigned to a work crew performing the modification and replacement of the control rod drives, and received the increment of excessive exposure, while engaged in the removal of a drive under the reactor vessel. In the performance of this specific job, the man was exposed to levels of radiation which ranged from 60 mr/hr to 800 mr/hr.

The following controls were in effect at the time of the incident: The area was restricted, a Radiation Work Permit (RWP) had been issued and the job was being supervised.

In retracing the incident to determine the cause of the exposure, the following information was determined:

1. The individual, employed by the contractor, arrived at Oyster Creek on Friday, April 27, 1973, was issued a film badge and attended an orientation course in Radiation Protection.

2. He was assigned to a crew scheduled to perform work within the scope of the control rod drive modification and replacement program. The work was conducted under the supervision of contractor personnel.

3. His total accumulated exposure through May 5, 1973 was 1210 mr as determined from film badge results. At this time, after reviewing his exposure, the individual was given permission to accumulate additional exposure to a level of 1700 mr, which was according to established guidelines.

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4. His total exposure on May 7, 1973 was 1615 mr (1210 mr film badge and 405 mr self-reading dosimeter) as recorded on the daily log sheet. At this time, the individual was assigned to a work crew scheduled to remove a control rod drive. The area in which the work was performed was adequately surveyed and the crew was under contractor supervision.

5. After performing the necessary drive work, the individual discovered that his self-reading dosimeters (200 mr, 500 mr and 1R) had all pegged upscale indicating an exposure in excess of 1 rem. The job had been performed in a high radiation area located under the reactor vessel.

6. His film badge was immediately processed and the results indicated 1810 mr for the period May 6 through May 8, 1973 inclusive, indicating the individual received approximately 1400 mr while performing the work.

After evaluation of the above information, the conclusion was reached that the cause of the overexposure was twofold; firstly, the failure of the individual of concern to periodically check his self-reading dosimeters to determine the amount of exposure he was receiving and, secondly, the failure of the contractor supervisor to, (being aware of the allowable exposure limits) periodically check the individual's exposure and to use more care in the assignment of work considering the man's previous accumulated exposure. Immediately upon discovering that the overexposure had occurred, a meeting was conducted between the contractor and Jersey Central Power & Light Company's staff to determine corrective action needed and to initiate measures of control to prevent recurrence of similar incidents. Corrective action taken involved the use of health physics personnel to more closely observe exposure of individuals engaged in work in Radiation Work Permit (RWP) areas. This was accomplished by having the health physics personnel perform the following:

1. Be aware of exposure limits for all contractor personnel requesting entrance to RWP areas prior to admittance.

2. Assure that all contractor personnel are informed as to the RWP requirements, are properly clothed, protected, monitored and record allowable exposure.

3. Monitor and record exposures of contractor personnel at least hourly, more frequently if required, and remove any individual from the area who reaches his allowable limit.

In addition, more stringent administrative requirements have been imposed on all contractor personnel to preclude the recurrence of this event. These requirements include daily meetings to discuss work to be performed in light of necessary radiation protection, the restriction from work in high radiation areas of all contractor personnel who receive an accumulated exposure of 2.0 rems, and the processing of film badges daily for all contractor personnel who are engaged in work in high radiation areas. It is felt that the above actions will assure Jersey Central Power & Light Company's management that a recurrence will not be experienced.

Jersey Central Power & Light Company had prepared and implemented radiological control of personnel engaged in work during the outage, through the establishment of administrative guidelines, the maintaining and reporting of all personnel exposure on a daily basis, and the orientation of all personnel in radiation protection. In addition, a supplemental system of memorandum writing was instituted to alert the contractor supervisors of personnel who were approaching pre-established limits. It is the feeling that Jersey Central Power & Light Company had maintained proper administrative control to prevent an occurrence of this nature and the reason for the incident was the failure of the contractor personnel involved to observe the rules and follow the proper safety practices.

We are enclosing forty (40) copies of this letter.

Very truly yours,

Donald A. Ross

Donald A. Ross
Manager, Nuclear Generating Stations

DAR:cs

Attachment

cc: Mr. J. P. O'Reilly, Director
Directorate of Regulatory Operations, Region I