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Jersey Central Power & Light Company

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August 15, 1973

RECEIVED AUG 16 1973 U.S. ATOMIC ENERGY COMMISSION Regulatory Mail Section

Mr. Robert J. Schemel
Directorate of Licensing
U. S. Atomic Energy Commission
Washington, DC 20545

References: (1) AEC Letter from R. J. Schemel, July 16, 1973.
(2) GE Topical Report, NEDM-10735, Supplement 6.
(3) JCP&L Co., Oyster Creek Nuclear Generating Station Facility Change Request Number 4 and Supplements.

Dear Mr. Schemel:

SUBJECT: FUEL DENSIFICATION

In your letter of July 16, 1973 (Reference 1), you requested that we provide the necessary analyses and other relevant data for determining the consequences of densification and the effects on normal operation, anticipated transients and accidents at the Oyster Creek Nuclear Generating Station (OC) using the AEC guidance attached to that letter. The letter stated that if analyses indicate that changes in design or operating conditions are necessary to maintain required margins as a result of using the Staff guidance, we should submit proposed changes and operating limitations with the analyses.

Reference 2 has been prepared by General Electric as a generic response to the AEC's densification concerns. JCP&L Co. has reviewed this document and finds it applicable to the fuel in OC. It should be noted, however, that the information contained in Table 4-1 associated with Plant A (which is OC) is not valid as it is representative of conditions not allowed by present Technical Specifications. A revised version of this information is presented in Table 1.

This response includes the analysis of both General Electric fuel (Types I and II) and EXXON fuel (Types III and IIIE) currently in the core. Analyses of the General Electric fuel specific to OC are contained in Reference 2 (Section 6.3), while the analyses of the EXXON fuel are contained in Attachment I of this letter. The results of all analyses are summarized in Table 1. Note that results have been provided in the postulated Loss of Coolant Accident (LOCA) case for three cases including:

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a. The current Interim Acceptance Criteria (IAC) model
b. The IAC model utilizing the AEC densification guidelines and applying the 95/90 confidence level to the gap conductance of the limiting rod
c. The IAC model utilizing the AEC densification guidelines and applying the 95/90 confidence level to the gap conductance for all rods.

The basis for these calculations and statistical confidence levels are discussed at length in Reference 2.

In addition to the requested analyses, a proposed Technical Specification is presented in Attachment II. This proposed Technical Specification allows for more direct monitoring of parameters determining the behavior of fuel under postulated LOCA conditions.

The assumed effects of fuel densification which have been considered in the analyses are the potential for (a) local power spikes resulting from axial fuel column gaps, (b) increased linear heat generation rate due to pellet axial shrinkage, (c) cladding collapse at the location of axial fuel column gaps, and (d) reduced pellet-clad thermal conductance due to increased pellet-to-clad gap as it may influence stored energy.

The results of conservative analyses of local power spikes are presented. The analyses yielded the conclusion that there is >95% confidence that no more than one rod in any existing fuel type will have a power spike >5% in magnitude. Further, due to the nature of the axial power distribution in a BWR, this maximum spike magnitude will not occur at the limiting or highest power generating axial location in the rod.

The results of the analysis of linear heat generation rate (LHGR) change due to densification show that the pellet axial shrinkage will be more than offset by the effects of axial thermal expansion in the rods near the limiting condition. Thus, no effect on LHGR is expected.

The results of analysis of cladding creep collapse for existing BWR fuel types with more than one cycle of operation are also summarized. The results show that creep collapse will not occur. Evaluations were calculated for existing fuel operating through September, 1974. Evaluations beyond this point will be conducted at a future date.

The additional major conclusions of the analyses are summarized below:

a. Normal Operation and Anticipated Transients:

The analysis of the assumed densification phenomena and their effect on plant safety bases have been examined. It is concluded that the safety design basis criteria and the current Technical Specification with its imposed total power peaking factor for both normal plant operation and anticipated transients are still applicable. Additional power peaking due to power spikes calculated utilizing the AEC model are adequately accounted for in conservation in existing calculations. Other assumed densification phenomena have minor effects on normal plant operation and on the outcome of anticipated transients.

b. Design Basis Accidents:

The four design basis accidents (Control Rod Drop Accident, Main Steam Line Break Accident, Refueling Accident, Loss of Coolant Accident) have been considered for effects due to the assumed densification phenomena. It has been concluded from these evaluations that the previously reported acceptable results for the Control Rod Drop Accident, Main Steam Line Break accident, and Refueling Accident are still valid.

Current analytical methods employed to evaluate Loss of Coolant Accidents have been utilized to determine the effects of assumed densification phenomena. Analysis of small break conditions indicate lower peak clad temperatures than previously reported. Therefore, current small break analysis remain valid.

The results of the design bases LOCA analyses (i.e., the recirculation line break) are significantly affected by the reduced gap conductance resulting from the utilization of the AEC model in representing the postulated densification phenomena. The effect is manifested in the calculated peak cladding temperature. The value of peak cladding temperature is calculated using both interpretations of the AEC densification model.

The utilization of either interpretation of the AEC densification model in conjunction with the proposed Technical Specification contained herein and additional constraints on the reactor power distribution requires the derating of the OC Nuclear Generating Station to meet the IAC. The utilization of a 95/90 confidence limit on the limiting rod results in an approximate derate of 4% from current licensed power. The utilization of a 95/90 confidence limit on all rods results in a derate of 11% from current licensed power. Analyses performed by both GE and EXXON, utilizing newer, more sophisticated models than either interpretation of the AEC densification model indicate no required derating.

With regard to the postulated LOCA, it is again emphasized that the OC Nuclear Generating Station satisfies the IAC now applicable to this event with no additional restrictions on modes of operation. It is made clear in Reference 2 and Attachment I that the principle effect of the Staff's fuel densification model is on the LOCA calculation and specifically a modification to the value of 1000 BTU/hr-ft^2 °F for pellet to clad gap conductance in the AEC approved General Electric ECCS evaluation models utilized by JCP&L Co. for OC fuel.

New, sophisticated analytical models representing the changes of gap conductance in relation to the various phenomena including fuel densification, associated with increasing exposure are in different stages of development. Such a model developed by EXXON Nuclear and expected to be submitted to the AEC in September, shows that adequate margin to limiting conditions now exist for both GE and EXXON fuel for the postulated design basis LOCA. General Electric is now developing a similarly sophisticated model which is expected to be submitted to the AEC by December 1, 1973. The more important effects considered by these models and not now represented in the AEC guidance are fuel pellet swelling, cracking and gap closure, and cladding creepdown. Information on these phenomena has already been presented to the Staff.

It is the judgment of JCP&L Co. based on a review of all evaluation models available and specifically the statistics which are the basis for the AEC guidance, that there is no safety reason for implementation of any additional operating restrictions, and that continued operation of OC under its existing license presents no undue hazard to the health and safety of the public.

Further, examination of the statistical analysis performed in Appendix A to Reference 2 establishes as suitable the conservatism of the AEC densification guidelines applying the 95/90 confidence level to the gap conductance of the limiting rod. However, it is the judgment of JCP&L Co. that even this interpretation of the available data is overly restrictive in light of the aforementioned results of more sophisticated analytical models and existing experimental data already presented to the AEC.

Finally, in response to the request for proposed operating limits consistent with the results of the analyses, the following recommendations are made:

a. It is recommended that a Technical Specification defined as a Limiting Condition for Operation and as described in Attachment II to this letter be implemented immediately. It should be emphasized that this recommendation provides more direct monitoring of LOCA limiting parameters and is consistent with present operating restrictions. Curve B of Figure 1 of Attachment II defines this limit.

b. In the event that the Commission finds it necessary to further restrict the operation of the OC reactor, it is recommended that this restriction be limited to that defined by the results of analyses utilizing the AEC densification guidelines and applying the 95/90 confidence level to the limiting rod. This restrictive mode of operation is shown for the GE fuel as Curve C of Figure 1 of Attachment II. Note that the Technical Specification proposed in a. above would be utilized but with Curve C as the limit line. Curves C' and C" representing the limit for the 95/90 confidence for the limiting rod assumption for the EXXON Type III and IIIE fuel will be supplied at a later date. Curves D, D' and D" represent the limiting conditions corresponding to the 95/90 confidence level on all rods for the GE Type II and EXXON Types III and IIIE fuel respectively.

c. It is recommended that the AEC continue its review of the fuel densification phenomenon and that the expected new analytical models for predicting the behavior of fuel under irradiation, including densification, and analyzing LOCA be reviewed and acted upon expeditiously.

We trust that the information contained herein meets the requirements specified in your letter of July 16, 1973. Please advise us promptly if any additional information is required.

Very truly yours,

Ivan R. Finfrock, Jr.

Ivan R. Finfrock, Jr.
Vice President

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Attachments

ATTACHMENT 1

EXXON NUCLEAR OYSTER CREEK DENSIFICATION ANALYSIS

(Compliance with AEC July 16, 1973, Guidelines)

The analyses discussed below reflect calculations performed in response to the letter to Jersey Central Power and Light from R. J. Schemel dated July 16, 1973. The guidelines given in Enclosure B to that letter for Exxon Nuclear fuel were followed in this analysis. The results of the analysis indicate that if the specified directives are applied, the peak power density of the Oyster Creek reactor must be restricted to an axial x radial peaking factor product of 2.146 while operating at full power (1930 MWth), or to a thermal power level reduced in inverse proportion to the axial x radial power peaking factor product ratioed to 2.33. Only analyses of the type III E fuel which are affected by the assumed densification phenomena are addressed below. Other analyses of the performance and safety aspects of this fuel are presented in Facility Change Request No. 4, dated January 18, 1973, and Supplements 1 and 3 thereto.

I. DENSIFICATION EFFECT ON NORMAL OPERATION

The maximum heating rate for the type III E fuel at rated power is as given in Supplement 1 to FCR No. 4 as 17.2 kw-ft with the reactor at full power. Under these conditions, the Minimum Critical Heat Flux Ratio was calculated to be 2.0 (XN-1 CHF correlation). The effects on these values of imposing the densification criteria stated in the AEC letter of July 16, 1973, are as follows:

A. Power Spike Model

Use of the power spike model presented in Supplement 3 to FCR No. 4 resulted in a conclusion that, with 95% confidence, no more than one fuel rod in the reactor would exceed a power spike (due to axial pellet column gaps) of about 2%. Utilizing this same model, but applying the following equation to determine the maximum gap size:

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

$$= \left(\frac{0.965 - 0.9405^{*}}{2} + .004 \right) L$$

$$= (.0162)\ L$$

The maximum axial gap size at a pellet column elevation of 126 inches (above which the fuel power is normally so low that the added spike effect can be ignored) is calculated to be 2.05 inches. Applying the same relative gap size distribution and gap frequency models indicated in Supplement 3 to FCR No. 4, it is calculated that, with 95% confidence, no more than one fuel rod in the reactor core would exceed a power spike of about 4.6%.

B. Linear Heat Generation Model

Utilizing the AEC criteria in the July 16, 1973, letter, the decrease in fuel column length (and conversely, the result of this effect on the LHGR) is calculated by:

$$\Delta L = - \left(\frac{0.965 - \rho i}{2} \right) L$$

$$= - .012\ L, \text{ or}$$

$$\frac{\Delta LHGR}{LHGR} = + .012$$

Compensating for this effect, but not accounted for in the original analysis for the type III E fuel, is the thermal expansion of the fuel

* Nominal pellet density (assembly average)

at power, which increases the fuel pellet axial length compared to the cold manufactured length. At a linear heat generation rate of 17.2 kw/ft, the axial thermal expansion of the fuel pellet column (utilizing the calculated UO_2 temperature at the inner edge of the pellet dish) is +1.2%, which is reflected as a reduction in LHGR below the design evaluation value of -1.2%. The engineering heat flux subfactor due to pellet density variation on an average assembly basis is:

$$\frac{\rho + \sigma}{\rho} = \frac{.9405 + .0055}{.9405} = 1.006$$

The calculation of the average and maximum LHGR in the reactor core depends upon (among other factors) the total active length of fuel pellet columns among which the thermal power load is shared. The specified length of the pellet columns in Exxon Nuclear fuel (cold) is 144 ± 0.25 inch, or ± .17%. All fuel rod pellet columns were verifed to fall within this manufacturing tolerance. It is assumed that the average length of all the fuel pellet columns is 144 inches, and no adjustment in the average or maximum LHGR is made to compensate for deviations from this mean.

C. Stored Energy Model

Applying the directive given in the AEC letter of July 16, 1973, the radial gap coefficient is calculated to be 500 BTU/hr^{o}F at the 95/90 confidence level based on the empirical gap coefficient correlation with the following attributes:

- LHGR = 17.2 kw/ft
- ρi = 93.5% of theoretical density
- 2σ = .234 inch (manufactured)

Applying this value and the UO_2 thermal conductivity data of Lyons et al, the calculated maximum pellet centerline temperature at 17.2 kw/ft is 4620^oF. This temperature is well below the 5080^oF melting point of UO_2 fuel. The effects of operational transients on the fuel centerline temperature is discussed in Section II below. The effects of fuel exposure and gadolinia burnable poison addition are as described in FCR No. 4, January 18, 1973.

D. Net Densification Effect on Normal Operation

The net effect of densification on normal operation of applying the assumed densification effects discussed above is to increase the assumed local peaking factor used to establish the maximum LHGR for core monitoring purposes. This increase is:

	Present Analysis	Previous Analysis
Power Spike	1.046	1.000
Decrease in column length due to densification	1.012	1.000
Fuel column thermal expansion	.988	1.000
Engineering heat flux factor	1.006	1.016
Total	1.052	1.016
Increase in hot spot factor in present analysis	1.030	

Thus, a peaking factor (F_D) of 1.036 should be superimposed on the core monitoring factors for the purpose of determining compliance with the Technical Specification limit of 17.2 kw/ft. As indicated in (C) above, this limit provides sufficient margin to insure operation below the UO_2 melting point and MCHFR limit during normal operation and anticipated transients. Maintaining the maximum LHGR at 17.2 kw/ft also maintains the calculated MCHFR at 2.0 for the same assumed core thermal hydraulic conditions given in FCR No. 4, January 18, 1973.

II. DENSIFICATION EFFECT ON TRANSIENT CONSEQUENCES

Sensitivity studies assuming undensified fuel indicate that, for the worst case of a transient involving a significant power spike (a turbine trip without bypass), the peak fuel pellet temperature increases about 100 °F. Considering the effects of assumed densification on the maximum steady-state UO_2 temperature indicated in Section I, C, above, a large margin to the UO_2 melting point remains during this transient.

Inspection of the results of the rod withdrawal incident analysis presented in FCR No. 4 shows that the APRM rod block setting will limit the local power density increase to less than 15%. This increase in the maximum steady-state peak power would increase the peak pellet temperature from about 4620 °F to 5200 °F, which exceeds the UO_2 melting point. To control this peak temperature to less than 5080 °F, the maximum local rod heat flux would have to be reduced about 3%. Since the results of the LOCA analysis (Section III) require a power reduction in excess of this value, the LOCA analysis results are controlling.

III. DENSIFICATION EFFECTS ON ACCIDENT CONSEQUENCES

The calculated consequences of the Main Steam Line Break accident, the Refueling accident, and the Rod Drop accident are not changed by the effects of the assumed densification guidelines. The effects of the AEC directives of the July 16, 1973, letter on the calculated Loss of Coolant accident consequences are as follows:

A. Power Spike Model

The effect of pellet column gaps on the results of the LOCA analysis has been considered. In the limiting case, two competing phenomena occur:

a. A pellet column gap in a given fuel rod results in an increase in the heat generation rate of adjacent rods. For BWR's, the worst case is a gap in a fuel rod adjacent to the rod which reaches the maximum clad temperature during the LOCA.

b. The presence of a pellet column gap in the axial plane of interest results in a reduction in bundle power by about 2% (the power fraction of the "missing" rod segment) for that axial plane.

Sensitivity studies of a limiting case, as described in (a) and (b) above, indicate that the net effect of pellet column gapping on the LOCA analysis is to slightly decrease the calculated peak clad temperature. Hence, this phenomena is disregarded in this LOCA analysis.

B. Linear Heat Generation Rate Model

Utilizing the AEC directives in the July 16, 1973, letter, the decrease in fuel length (core average*) for the purpose of LOCA calculations is obtained from:

$$\Delta L = -\left(\frac{0.965 - \rho i}{2}\right) L$$

The mean ρi for this type III E fuel was reported in Supplement 3 to FCR No. 4 as 94.05% TD. Hence, $\Delta L/L = -.01225$.

The core averaged fuel pellet thermal expension is calculated to be about 0.5% at a core-averaged fuel power of 5.2 kw/ft. Hence, the net effect of densification on the LHGR for the purpose of LOCA calculations is + .012 - .005 = + .007 or + .07%.

C. Stored Energy Model

The AEC directive of July 16, 1973, to utilize reported gap coefficient values to obtain a gap coefficient model as a function of LHGR, gap size, and pellet diameter that predicts the data with 95% confidence that 90% of future events will exceed predictions was followed. Table (1) is a summary of all applicable gap coefficient data at beginning of life with helium fill gas used to construct the gap coefficient model. The observed gap heat transfer coefficients, pellet diameters, and gap-to-diameter ratio used in the model development are presented in Table (2). The form of the gap coefficient model assumed to describe the data was based on the form of the analytical solution for heat transfer with cylindrical geometry. The empirical model has the form:

$$h_g = a_0(LHGR)^{a_1} (g/D)^{a_2} (D)^{a_3}$$

* Assuming the core is loaded with type III E fuel.

where

h_g = gap heat transfer coefficient, Btu/hr.-ft^2-°F

LHGR = linear heat generation rate, kw/ft

g/D = cold diametrial gap-to-pellet diameter ratio

The parameters of the empirical model representing the mean of the data were evaluated by a least square analysis from Table II.

Since the AEC directive specified that a 95/90 lower tolerance limit accompany the gap coefficient model, a statistical evaluation of the difference between the predicted and the reported gap coefficient was performed. It was found that multiplication of the empirical gap coefficient for the mean of the data by 0.826 yields a predicted gap coefficient value that satisfies the 95/90 lower tolerance limit criteria. The resultant gap coefficient model for 95% confidence that 90% of future events will exceed prediction is

$$h_g = 14.98\ (LHGR)^{0.374}\ (g/D)^{-0.523}\ (D)^{-0.94}.$$

Figure 1 provides a comparison of the prediction with the 95/90 empirical gap coefficient model and the reported gap coefficients. All of the reported gap coefficients except a single point from Reference (4) at 7.5 kw/ft, a g/D = 0.0326 and reported gap coefficient of 473 Btu/hr.-ft^2-°F are shown to be underpredicted in Figure 1.

Instantaneous densification of the UO_2 fuel pellets in the type III E fuel is assumed. Since the LOCA calculation results are sensitive to the bundle-averaged stored energy, the pellet diameters and gap sizes are calculated assuming the lowest mean value of the individual pellet lot density is used to determine the densified pellet diameter. As reported in Supplement 3 to FCR No. 4, the lowest mean density value of an individual pellet lot was 92.3% TD. Hence,

$$\Delta r/r = - \left(\frac{0.965 - 0.923}{3}\right) = - 0.014$$

The cold fuel pellet OD is adjusted using this value and a "densified" pellet-to-clad cold gap obtained. The curves derived from fits to the experimental data described above were used in conjunction with the "densified" cold gaps to obtain gap conductivities for each fuel rod. These values are then used to calculate the peak clad temperature during the LOCA as described in FCR No. 3, January 13, 1973. The results of this calculation are presented in Figure 2, where the peak clad temperature is plotted as a function of the product of the axial x radial peaking factors, with the reactor power assumed to be 1930 MWth. This figure indicates that the peak clad temperature is 2300 °F when the axial x radial product is 2.146. The range of gap coefficients for several values of this product is as shown on the figure.

The above calculation conservatively assumes maximum densification to occur instantly although it is expected that full densification would not be accomplished until several days or weeks have elapsed. During this period gap closure as a result of pellet cracking would tend to compensate for a reduction in gap coefficient due to densification.

REFERENCES

1. A. S. Bain, *Microscopic, Autoradiographic and Fuel/Sheath Heat Transfer Studies on UO_2 Fuel Elements*, AECC-2588, June 1966.

2. R. N. Duncan, *Rabbit Capsule Irradiation of UO_2, Terminal Report*, CVNA-142, 1962.

3. G. Kjaerheim and E. Rolstad, *In-Pile Determination of UO_2 Thermal Conductivity, Density Effects and Gap Conductance*, HPR-80.

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

Table 1

SUMMARY OF PHYSICAL PARAMETERS OF DATA USED

DEVELOPMENT OF EMPIRICAL GAP COEFFICIENT MODEL

	AECL-2588(1)	CVNA-142(2)	HPR-80(3)	NEDM-10735(4)
Pellet Diameter, in	∿ .65	∿ .43	.49	.488
Diametral gap, in	.025	.0256	.0066	.016
Linear Heat Generation Rate, kw/ft	25.3	18,24	2.8 - 15.0	17.5 - 20
Exposure, MWD/MTM	∿ 0	∿ 0	∿ 0	∿ 0
Fill Gas	He	He	He	He
Reference Temperature	Melting	Equiaxed	Thermocouple	Equiaxed
Cladding Material	SST	Zr	Zr	Zr
% Theoretical Density	97.3	94	96	95.7
External Pressure, psi	100	--	406	∿ 1000
Cladding Thickness, in	∿ .026	--	.0218	0.030

Table 2

SUMMARY OF DATA USED FOR

DEVELOPMENT OF EMPIRICAL GAP COEFFICIENT MODEL

Gap Coefficient; Btu/hr-ft^2-°F	LHGR kw/ft	g/D	Pellet O.D., inch	Data Source
705	17.5	.0326	.488	GE-NEMD 10735
550	17.5	.0326	.488	GE-NEDM 10735
555	17.5	.0326	.488	GE-NEDM 10735
473	17.5	.0326	.488	GE-NEDM 10735
1010	20.0	.0326	.488	GE-NEDM 10735
623	20.0	.0326	.488	GE-NEDM 10735
486	25.3	.0404	.650	AECL-2588
486	25.4	.0404	.650	AECL-2588
570	18.0	.0590	.430	CVNA-142
520	24.0	.0590	.430	CVNA-142
565	2.83	.0133	.490	HPR-80
580	3.66	.0133	.490	HPR-80
600	4.57	.0133	.490	HPR-80
625	5.76	.0133	.490	HPR-80
700	7.99	.0133	.490	HPR-80
720	8.72	.0133	.490	HPR-80
740	8.99	.0133	.490	HPR-80
750	9.17	.0133	.490	HPR-80
760	9.57	.0133	.490	HPR-80
830	11.37	.0133	.490	HPR-80
1050	14.97	.0133	.490	HPR-80
600	4.48	0.135	.490	HPR-80
620	5.36	0.135	.490	HPR-80
630	5.70	.0135	.490	HPR-80
670	7.16	.0135	.490	HPR-80
720	8.56	.0135	.490	HPR-80
730	8.84	.0135	.490	HPR-80
750	9.39	.0135	.490	HPR-80
755	9.54	.0135	.490	HPR-80
800	10.55	.0135	.490	HPR-80
810	10.79	.0135	.490	HPR-80
860	12.04	.0135	.490	HPR-80
880	12.47	.0135	.490	HPR-80

Table 2 (Continued)

910	13.11	.0135	.490	HPR-80
940	13.50	.0135	.490	HPR-80
970	14.11	.0135	.490	HPR-80

FIGURE 1

COMPARISON OF PREDICTED GAP COEFFICIENTS
VERSUS REPORTED GAP COEFFICIENTS

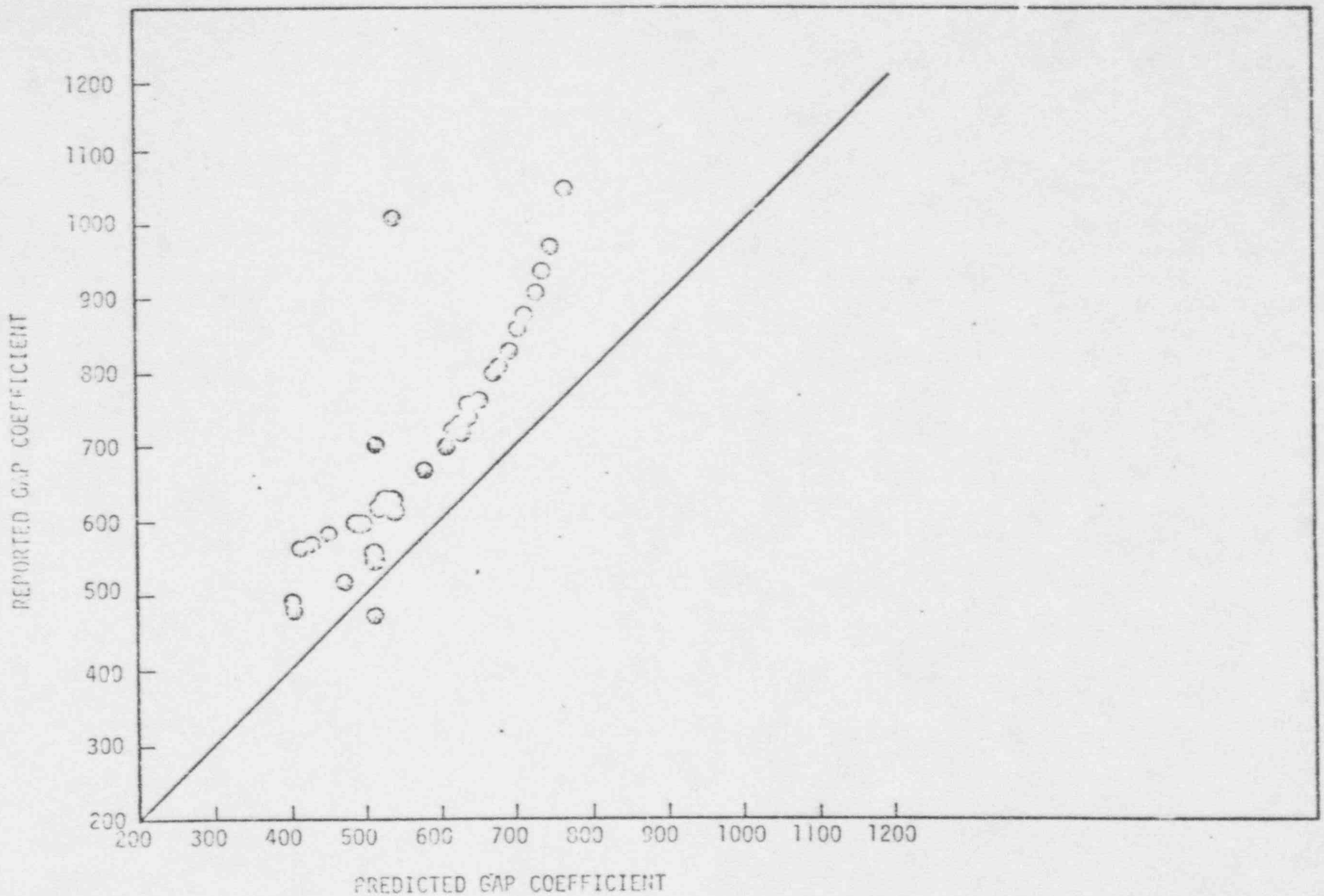
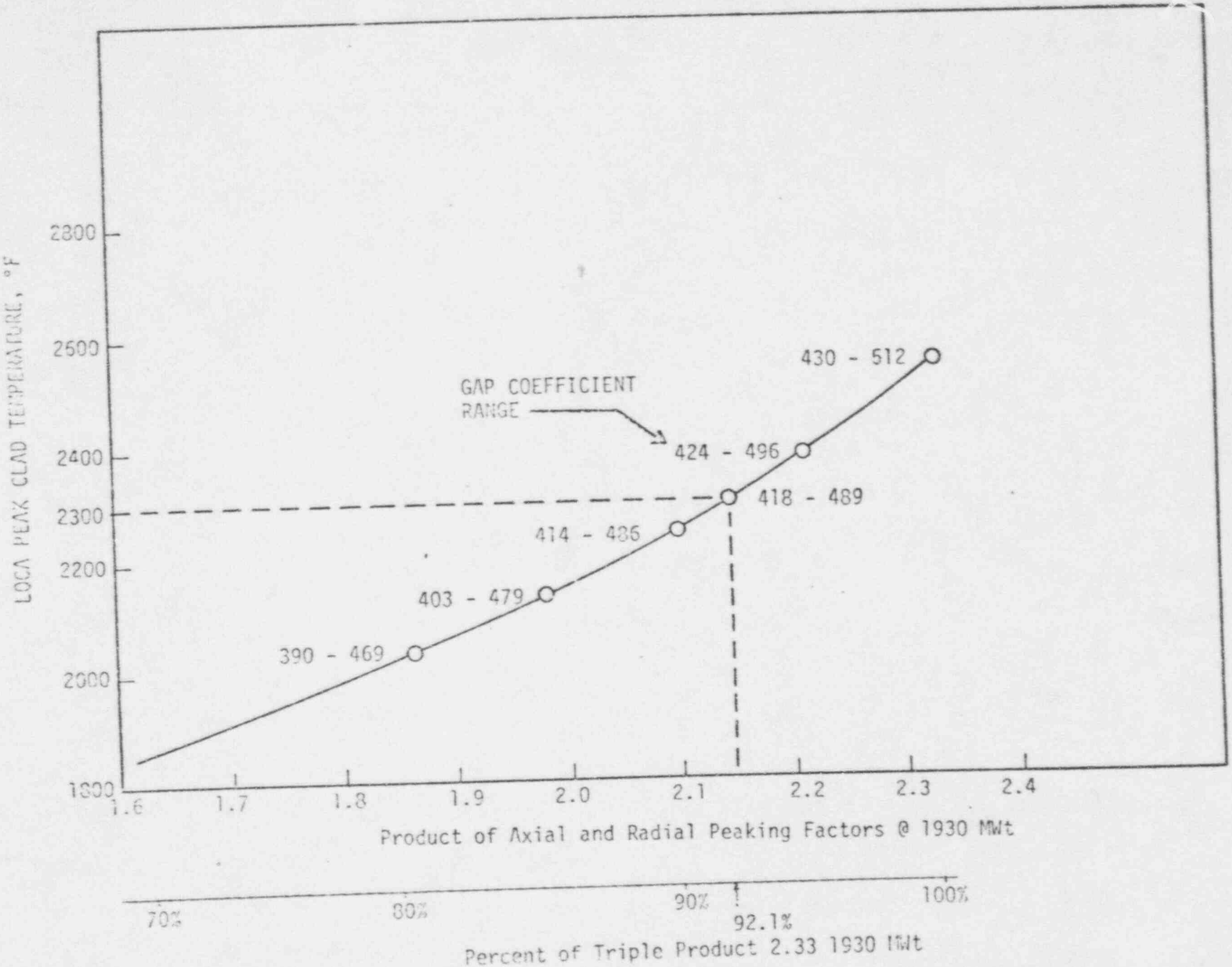


FIGURE 2

OYSTER CREEK TYPE IIIE FUEL

Per AEC Letter to GPU, July 16, 1973



APPENDIX A

RESULTS OF EXXON ANALYSIS
FOR TYPE III FUEL

CURRENT ANALYSIS:

hgap	PCT	APLHGR	RxA	Power
1000	2225	13.3	2.33	1930

Using AEC Guidance of July 16, 1973:

410-475	2601	13.3	2.33	1930
400-456	2300	12.2	2.20	1850

Power Spike Analysis — See Discussion for IIIE

Transient Analysis — See Discussion for IIIE

ATTACHMENT II

PROPOSED TECHNICAL SPECIFICATION CHANGE

1A Specification to be Changed

Section 3. Limiting Conditions for Operation

1B Extent of Change

Add Specification 3.10 to Section 3.

Add Figure 3.10.1

1C Change Requested

3.10 AVERAGE PLANAR HEAT GENERATION RATE

Applicability: Applies to the monitoring of the maximum average planar linear heat generation rate (MAPLHGR)

Objective: To limit the APLHGR in such a manner as to conform to the peak clad temperature limitations during a postulated loss-of-coolant accident as specified in the Interim Acceptance Criteria.

Specification: The average linear heat generation rate at any axial cross section of any fuel bundle in the core (Average Planar Linear Heat Generation Rate, APLHGR) shall not exceed the operating level (MAPLHGR) shown by Curve B at Figure 3.10.1

(See Attached Figure 3.10.1)

Bases: To be provided subsequent to AEC Staff evaluation.

(Note: The justification for choosing Curve B of figure 3.10.1 is given in the cover letter to this submittal).

2A Specification to be Changed

Section 4. Surveillance Requirements

2B Extent of Change

Add Specification 4.10 to Section 4.

2C Change Requested

4.10 AVERAGE PLANAR HEAT GENERATION RATE

Applicability: Applies to the surveillance of the average Planar Linear Heat Generation Rate (APLHGR).

Objective: To assure that the APLHGR is within the limitations imposed by Curve B of Figure 3.10.1

Specification: Daily during reactor operation, the maximum Average Planar Linear Heat Generation Rate shall be estimated and checked against Curve B of Figure 3.10.1 and adjusted if required.

Basis: The peak clad temperature which may result in the event of a postulated loss-of-coolant accident varies proportionally to the MAPLHGR. Daily surveillance of the MAPLHGR will assure that the LOCA peak clad temperature will conform to the limitations imposed by the Interim Acceptance Criteria.