MAY 25 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

Change No. 15
License No. DPR-16

Gentlemen:

By letters dated January 18, 1973, February 22, 1973, May 22, 1973, and May 23, 1973, you submitted Facility Change Request No. 4 and its supplements which described and provided supporting analyses and information for the use in Oyster Creek Cycle 3 of Type III E fuel assemblies manufactured by Exxon Nuclear Corporation. In addition, on May 10, 1973, you submitted Change Request No. 15 to the Oyster Creek Technical Specifications that requested authorization to use the XN-1 Critical Heat Flux Correlation for the determination of safety limits for operation of the Oyster Creek Nuclear Generating Station.

We have reviewed the above submittals and have concluded, subject to the conditions stated below, that operation with the Cycle 3 core, as described in Facility Change Request No. 4 and its Supplements No. 1, 3, and 4, and implementation of Technical Specification Change Request No. 15, modified as discussed below, do not present a significant hazards consideration and that there is reasonable assurance that the health and safety of the public will not be endangered.

Since the worst case of the loss-of-coolant accident (LOCA) analysis is reported to occur at 10,000 MWD/MT with an axial x radial peaking factor of 2.33, Technical Specification Change Request No. 15 has been modified to include a limit on the axial x radial peaking factor. This was accomplished by adding a note to Figure 2.1.1 specifying that the local peaking factor must always be assumed to be 1.29. This modification has been discussed with your staff.

Until you supply additional information providing details on the effect of radial and local peaking factor variations on the LOCA analysis for exposures greater than 10,000 MWD/MT, the exposure of Type III E assemblies shall be limited to 10,000 MWD/MT.

Until the effects of an inert rod within the fuel assembly on the spray heat transfer coefficient for the LOCA analysis can be accounted for to our satisfaction, we conclude that the calculated 2200°F peak clad temperature following a postulated LOCA provides an appropriate margin to the 2300°F limit of the Interim Policy Statement dated July 15, 1971.

This authorization in no way affects the generic evaluations presently being performed in regard to fuel densification and control rod drop accident analyses. The refueling with Type III E fuel does not introduce new safety considerations in these generic evaluations.

Accordingly, pursuant to 10 CFR Part 50, Section 50.59, you are hereby authorized to operate the Oyster Creek Plant with the Cycle 3 core, as described in Facility Change Request No. 4 and its Supplements Nos. 1, 3, and 4, and the Technical Specifications of Provisional Operating License No. DPR-16 are hereby changed as indicated in Attachment A.

Sincerely,

Donald J. Skovholt
Assistant Director for
Operating Reactors
Directorate of Licensing

Enclosures:
1. Attachment A - Change No. 15 to the Technical Specifications
2. Safety Evaluation

cc w/enclosures:
see next page

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ATTACHMENT A

CHANGE NO. 15 TO THE TECHNICAL SPECIFICATIONS

PROVISIONAL OPERATING LICENSE NO. DPR-16

JERSEY CENTRAL POWER & LIGHT COMPANY

DOCKET NO. 50-219

1. Replace page 2.1-2 in its entirety with the following:

"A critical heat flux occurrence results in a decrease in heat transferred from the clad and, therefore, high clad temperatures and the possibility of clad failure. However, the existence of a critical heat flux occurrence is not a directly observable parameter in an operating reactor. Furthermore, the critical heat flux correlation data which relates observable parameters to the critical heat flux magnitude is statistical in nature.

"The safety limit curves shown in Figure 2.1.1 represent the conditions for which there is 99 percent confidence that the most limiting rod has a minimum critical heat flux ratio (MCHFR) greater than 1.0. The MCHFR value was determined using the design basis critical heat flux correlation given in JN-72-18 (1). The operating range with MCHFR ≥1.0 is below and to the right of these curves.

"The design basis critical heat flux correlation is based on an interrelationship of reactor coolant flow and steam quality. Steam quality is determined by reactor power, pressure, and coolant inlet enthalpy which in turn is a function of feedwater temperature and water level. This correlation is based upon experimental data taken over the entire pressure range of interest in a BWR, and the correlating line was determined by the statistical mean of the experimental data.

"Curves are presented for two different pressures in Figure 2.1.1. The upper curve is based on nominal operating pressure of 1035 psia. The lower curve is based on a pressure of 1250 psia. In no case is reactor pressure ever expected to exceed 1250 psia because of protection system settings well below this value and, therefore, the curves will cover all operating conditions with interpolation. For pressures between 600 psia (the lower end of the critical heat flux correlation data) and 1035 psia, the upper curve is applicable with increased margin."

2. Page 2.1-3, first paragraph, change the total peaking of 3.03 to 3.01 wherever it appears in the paragraph (lines 2, 11, 15, and last line).

3. Page 2.1-5, change reference (1) to read as follows:

   "(1) L. H. Steves, A. M. Sutey, O. E. Fitzsimmons, "XN-1 Critical Heat Flux Correlation for Boiling Water Reactor Fuel".

4. Page 2.1-6, revise Figure 2.1.1, Fuel Cladding Integrity Safety Limit.

5. Page 2.3-1 and 2.3-2, Limiting Safety System Settings, change total peaking factor of 3.03 to 3.01 (sections 1a and 2a).

6. Page 2.3-4, paragraph 3, lines 2, 8, and 9, change 3.03 to 3.01.

SAFETY EVALUATION BY DIRECTORATE OF LICENSING

JERSEY CENTRAL POWER & LIGHT COMPANY

OYSTER CREEK NUCLEAR GENERATING STATION UNIT 1

DOCKET NO. 50-219

Introduction

By application dated January 18, 1973, Jersey Central Power & Light Company submitted Facility Change Request No. 4 (FCR 4) that described the Type III B fuel assembly manufactured by Exxon Nuclear Corporation and proposed to load 148 of the Type III B assemblies into the Oyster Creek Nuclear Plant core for operation in Cycle 3. Supplement No. 1 to FCR 4, submitted by letter dated February 22, 1973, described modifications made to the Type III B fuel assembly and presents the effects these modifications have on the analyses presented in FCR 4. The modified assembly is designated Type III E.

The original Oyster Creek core contained 560 assemblies, designated Type I, manufactured by General Electric Company. These assemblies contained no gadolinia. Poison curtains were used for supplementary control. In the fall of 1971, a partial reload was performed and twenty-four (24) fuel assemblies containing gadolinia, manufactured by General Electric Company and designated Type II, were loaded. The Type II assemblies were the subject of Facility Change Request No. 1 and were approved for use in Oyster Creek by letter dated September 22, 1971. In the spring of 1972, the reload for Cycle 2 operation consisted of 132 Type II assemblies and 4 Type III assemblies manufactured by Exxon Nuclear Corporation. The Cycle 2 reload was approved by letter dated June 12, 1972, and the Type III fuel assembly was approved by letter dated May 18, 1972. Type III B fuel assembly was not built and, therefore, will not be addressed in this evaluation. Types II, III, and III E incorporate minor modifications; but each type is basically similar to the original Type I design, the most significant modification being the incorporation of gadolinia bearing rods in the assembly. A comparison of the mechanical and thermal characteristics of all fuel types used in the Oyster Creek core is shown in Table I. The loading of 148 assemblies of Type III E fuel was approved by letter dated April 16, 1973.

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Evaluation

The differences between the Type III E fuel assembly and the Type III are as follows:

a. There are four rods, rather than two, containing 1 w/o gadolinia.

b. The center spacer-capture rod which was made up of segments containing enriched UO_2 fuel has been replaced with a spacer-capture rod filled with solid zircaloy 2.

c. Nineteen centrally located rods have had their initial theoretical density increased from 93.5 to 94.5 percent.

d. The same nineteen centrally located rods have had their fuel pellet diameters increased 1 to 3 mils, thus, reducing the diametral gap between fuel pellet and cladding.

A comparison of the infinite multiplication factors for all four type fuel assemblies in Oyster Creek are given in Table II. The values for Type III E fuel fall between those of Type I and Type II fuel.

A comparison of the core nuclear characteristics for Cycles 1, 2, and 3 are given in Table III. The values used in the transient analyses for the 1930 MWt application (Amendment 65) and the reanalysis of the pressurization transients (Amendment 69) are also listed. The effective multiplication factors for the Cycle 3 core loading are higher than for the two previous cycles. The Cycle 3 reload consists of 148 Type III E assemblies, whereas, the Cycle 2 reload consisted of 132 Type II and 4 Type III assemblies. However, the shutdown margin design criterion of k_{eff} less than 0.99 with the rod of most reactivity worth fully withdrawn is still met. The minimum shutdown margin occurs at 2000 - 2500 MWD/MT due to burnup of the gadolinium poison and is greater than the Technical Specification requirement.

The nuclear parameters for Cycle 3 were compared with those used in the analyses reported in Amendments 65 and 69 in conjunction with the results of the parametric studies reported in FCR 4. We have also reviewed the transient analyses of the single rod withdrawal error presented in Supplement No. 1 to FCR 4. We conclude that the analyses satisfactorily represent the Cycle 3 core conditions and the results are acceptable.

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TABLE I

Characteristics	Type I	Type II	Type III	Type III E
Fuel Materials	UO_2	UO_2	UO_2	UO_2
Total Rods/Bundle	49	49	49	49
Geometry	7x7	7x7	7x7	7x7
Standard Rods/Bundle	48	44	46	44
Spacer Capture Rods/Bundle	1	1	1	1
Poison Rods/Bundle	0	4	2	4
Unfueled Rods/Bundle	0	0	0	1
Rod Pitch	0.738	0.738	0.738	0.738
Water to Fuel Volume Ratio	2.38	2.47	2.43	2.48
Initial Enrichment w/o Fissile Isotope				
Low U-235	1.9 (3 rods)	1.40 (1 rod)	1.59 (5 rods)	1.59 (5 rods)
Medium Low U-235	0 rods	1.83 (6 rods)	0 rods	0 rods
Medium U-235	1.67 (16 rods)	2.24 (10 rods)	2.42 (12 rods)	2.42 (12 rods)
High U-235	2.44 (30 rods)	2.94 (32 rods)	2.87 (32 rods)	2.87 (31 rods)
Bundle Average Fissile	2.11	2.63	2.63	2.63
Pellet Dish, % of Undished Volume	2.0 (in some rods)	2.0 (in some rods)	2.0 (in all rods)	2.0 (in all rods)
Average Pellet Density, % Theoretical Density	94	94	93.5	94.5 (19 rods) 93.5 (29 rods)

TABLE I (cont.)

Characteristics	Type I	Type II	Type III	Type III E
Number of Rods and Pellet Diameter (inches)	All - 0.488	All - 0.487	27 - 0.488 22 - 0.468	8 - 0.491 11 - 0.489 7 - 0.488 22 - 0.468
Cladding Material	Zr-2	Zr-2	Zr-2	Zr-2
Cladding O.D. (inches)	0.570	0.563	0.570	0.570
Number of Rods and Clad Wall Thickness (inches)	All - 0.0355	All - 0.032	27 - 0.0355 22 - 0.3455	27 - 0.0355 22 - 0.0455
Active Fuel Length (inches)	144	144	144	144
Gas Plenum Length (inches)	11 -1/4	11-1/4	10-5/8	10-5/8
Fill Gas	Helium	Helium	Helium	Helium
Total Fuel Length, ft/bundle	586.9	586.9	586.9	586.9
Heat Transfer Surface, ft^2/bundle	87.6	86.5	87.6	85.98
Bare Rod Flow Area, ft^2/bundle	.1057	.1078	.1057	.1057
Tech Spec Limit Total Peaking Factor	3.03	3.03	3.03	3.01
Max. Heating Rate, kW/ft	17.2	17.2	17.2	17.2
Max. Heat Flux, Btu/hr-ft^2	393,400	398,300	393,400	393,400
Corresponding Clad Surface Temperature, °F	568	568	568	568

TABLE I (cont.)

Characteristics	Type I	Type II	Type III	Type III E
Corresponding CHFR	1.90 Hench Levy	1.90 Hench Levy	1.90 Hench Levy	2.0 XN-1
Fuel Supplier	GE	GE	EXXON	EXXON
Number of Assemblies in Cycle 1	560 'til fall '71 536 'til spring '72	0 'til fall '71 24 between fall '71 & spring '72	0	0
Number of Assemblies in Cycle 2	400	156	4	0
Number of Assemblies in Cycle 3	252	156	4	148

Table II

	Type I*	Type II**	Type III**	Type III E**
K_∞ 68°F	1.128	1.162	1.221	1.145
K_∞ 300°F	1.129	1.159	1.224	1.139
K_∞ 549°F	1.128	1.156	1.224	1.132
K_∞ Full Power, 0% Void, Hot	1.122	1.150	1.219	1.127
K_∞ Full Power, 32% Void, Hot	1.100	1.135	1.203	1.111
K_∞ Full Power, 64% Void, Hot	1.061	1.106	1.171	1.081
$\Delta K_\infty/K_\infty$ Doppler Defect (Hot Standby to Full Power)	-0.0048	-0.0047	-0.0046	-0.0045
$\Delta K_\infty/K_\infty$ Void Defect (0 to 32% Voids)	-0.0197	-0.0135	-0.0130	-0.0141
$\Delta K_\infty/K_\infty$ Temp. Defect (68°F to 549°F)	-0.0004	-0.0058	+0.0024	-0.0114
$\Delta K_\infty/K_\infty$ Control Rod Worth, Cold	-0.153	-0.150	-0.163	-0.151

* With poison curtains
** With gadolinia rods

Table III

	BOC 1	BOC 2	At Max. Reactivity C 2	EOC 2	BOC 3	At Max. Reactivity C 3	EOC 3	Amdt 65 Trans Anlys	Amdt 69 Trans Anlys
K_{eff} (Cold -68°F)									
Uncontrolled	1.13	1.105	1.112	-	1.128	1.137	-	-	-
Fully Controlled	0.96	0.943	0.949	-	0.964	0.970	-	-	-
Strongest Single Rod Out	0.98	0.977	0.991	-	0.984	0.987	-	-	-
Total Control Rod Worth, Δk	- 0.17	-0.162	-0.163	-	-0.164	-0.167	-	-	-
β, Delayed Neutron	0.007	0.0056	-	0.0049	0.0056	-	0.0049	0.00643	0.00547
ℓ^*, Neutron Life-time, msec	-	40.6	-	40.2	40.6	-	39.7	39.2	33.4
Void Coefficient @≈ 33% voids									
$\Delta k/k\%$ Void x 10^4	-13.65	-10.1	-	-9.3	-11.0	-	-9.4	-15.65	-11.47
\$/% Void/initial Void%	-6.44	-5.95	-	-6.23	-6.45	-	-6.30	-7.96	-6.84
Doppler Coefficient @≈ 33% Voids									
$\Delta k/k/°F$ x 10^5	-1.23	-1.09	-	-1.16	-1.08	-	-1.15	-1.19	-1.39
¢/°F	-0.176	-0.195	-	-0.235	-0.192	-	-0.234	-0.166	-0.254

We expect no significant difference in the thermal and hydrualic performance of Type III E fuel as compared to the other fuel types in the core. By letter dated May 10, 1973, the licensee submitted Technical Specification Change Request No. 15 that would permit the use of the XN-1 Critical Heat Flux Correlation as the basis for the Safety Limits and Limiting Safety System Settings. We reviewed the Exxon Nuclear Corporation report, "XN-1 Critical Heat Flux Correlation for Boiling Water Reactor Fuel", JN-72-18, August 1, 1972. The review consisted of: (1) comparison with experimental data presently available in the Regulatory staff's data bank; (2) comparison with the existing BWR critical heat flux correlations; and (3) a review of the Oyster Creek reload fuel geometry to establish applicability of the correlations.

All data points from the Exxon Nuclear tests, as well as data points from other sources, fell above a line representing the XN-1 correlation multiplied by 0.75. This means that all data points exhibited a higher critical heat flux than would be predicted by 0.75 x the XN-1 correlation value for the same conditions. Using statistical methods for analyzing their data base, Exxon concluded that with 99 percent confidence at least 99 percent of the critical heat flux values exceed 0.75 x XN-1 correlation values for the same conditions. The licensee, therefore, proposed that 0.75 x the XN-1 critical heat flux value be the new Safety Limit. No change to the Limiting Safety System Settings is proposed.

From our review of the fuel geometry, both the General Electric fuel and the Exxon fuel, and based on the data comparison and correlation comparison discussed above, we conclude that the use of the XN-1 correlation as a basis for the Safety Limits and Limiting Safety System Settings of the Oyster Creek reactor is acceptable.

Attachment 1 to FCR 4 consists of the "Summary of the Quality Assurance Program for Oyster Creek Station Reload Fuel" and its Appendices: "A - Summary of Design and Quality Assurance Activity of GPUSC", "B - Some Representative Areas Evaluated During the Audits of Fuel Assembly Manufacture", and "C - Quality Control Manual - Jersey Nuclear Corporation". The staff reviewed these documents and held a meeting

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with personnel from General Public Utilities Service Corporation (GPUSC) and Exxon Nuclear Corporation (previously Jersey Nuclear Corporation) on March 21, 1973, to discuss clarification of some items in the submittal. The items discussed were: documentation of disposition of nonconforming items, heat treatment of tubing, moisture control, tubing scratch limits, gadolinia process control, and inspections of tubing. Based on our review of the above documents and the clarifying discussion with Exxon and GPUSC, we conclude that the Quality Assurance Program being implemented by Exxon and the licensee for the Oyster Creek Station reload fuel is acceptable.

Two of the accident analyses required renalysis for Cycle 3. These were the Loss of Coolant Accident and the Control Rod Drop Accident. The LOCA analytical methods and input parameters were described in FCR 4 and the results given in FCR 4 were applicable to a Type III B fuel assembly which will not be part of this reload. Supplement No. 1 to FCR 4 presents the results for Type III E fuel assembly, and Supplement No. 3 to FCR 4, submitted by letter dated April 17, 1973, presents in answer to our Question 9 of Attachment B to our letter dated April 3, 1973, the differences between Exxon's MOXXY code used for the heatup calculation and the AEC's MOXY code.

The blowdown analysis presented in the Oyster Creek FDSAR and resulting transient heat transfer coefficients given in Amendment 67 were used for the LOCA analysis of the reload fuel. The licensee also performed an independent blowdown analysis using RELAP-3 and reported that the RELAP analysis confirmed the previously presented depressurization transient. The reload fuel is similar to the rest of the fuel in the core in terms of heat generation and heat transfer characteristics with one exception - the use of an inert rod at the center of the fuel assembly. Since the calculated peak clad temperature is only 2200°F, there is reasonable assurance that the peak clad temperature will stay below 2300°F taking into account any effect of the inert rod on the spray heat transfer coefficients.

The differences between the MOXXY code and the AEC developed MOXY code are minor. The MOXXY code predicts approximately the same temperature transient for a given case as does GE's core heatup code. This has been demonstrated in Figure 17 of FCR 4.

Therefore, this analysis is concluded to be an appropriate up-dating of the analysis presented in Amendment 67 and the results continue to satisfy the criteria of the Interim Policy Statement dated July 15, 1971. The effects of fuel densification on the LOCA analysis is presently under evaluation on a generic basis (i.e., review of report,

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NEDM-10735, "Densification Considerations in BWR Fuel Design and Performance"). However, the use of Type III E fuel in this refueling is not expected to change the results of the fuel densification evaluation.

Since the LOCA analysis assumes an axial x radial peaking factor of 2.33 with the flatter local distribution at 10,000 MWD/MT, the total peaking factor for the limiting case of the LOCA analysis is less than the Technical Specification limit of 3.01, and it is necessary to limit the axial x radial peaking factor. We are, therefore, modifying the proposed Technical Specification Change No. 15 to include on Figure 2.1.1 the limitation: "The local peaking factor is always assumed to be 1.29."

Since the local peaking factors in the central rods of the fuel assembly continue to increase with exposure after 10,000 MWD/MT, it has not been shown that the 10,000 MWD/MT case is the worst LOCA case. Therefore, until the licensee submits additional information in regards to the effect of radial and local peaking factors on the LOCA analysis for exposures greater than 10,000 MWD/MT, the exposure of the Type III E assemblies will be limited to 10,000 MWD/MT.

The Control Rod Drop Accident analysis presented by the licensee was reviewed and the results are similar to those being evaluated presently on a generic basis for General Electric Boiling Water Reactors. It is anticipated that as a result of this generic evaluation there will be some changes to the Technical Specifications in regards to the design limit for maximum control rod worth and in regards to requirements to assure that these limits will not be exceeded.

Conclusions

Based on the above considerations, we have concluded that operation with the Cycle 3 core, as described in Facility Change Request No. 4 and its Supplements Nos. 1, 3, and 4, and implementation of the changes to the Technical Specifications, as proposed in Change Request No. 15 modified as described above, do not present a significant hazards consideration and that there is reasonable assurance that the health and safety of the public will not be endangered. This conclusion does not affect any subsequent action required by the results of the generic fuel densification evaluation.

T. V. Wambach
Operating Reactors Branch #1
Directorate of Licensing

Robert J. Schemel, Chief
Operating Reactors Branch #1
Directorate of Licensing

~~Date:~~ MAY 25 1973

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