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Docket No. 50-219

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

Gentlemen:

The Commission has issued the enclosed Amendment No. 24 to Provisional Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station. This amendment consists of changes to the Technical Specifications and is in response to your application dated June 23, 1977.

The amendment modifies the Technical Specifications to delete specifications and bases applicable to fuel types I and III which are no longer in the Oyster Creek core and to extend the MAPLHGR curves for the fuel types still in the core.

Copies of the related Safety Evaluation and the Federal Register Notice also are enclosed.

Sincerely,

Original signed by
George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosures:
1. Amendment No. 24 to License DPR-16
2. Safety Evaluation
3. Federal Register Notice

cc w/enclosures:
see next page

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JERSEY CENTRAL POWER & LIGHT COMPANY

DOCKET NO. 50-219

OYSTER CREEK NUCLEAR GENERATING STATION, UNIT NO. 1

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 24
License No. DPR-16

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The applications for amendment by Jersey Central Power and Light Company (the licensee) dated June 23, 1977, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B. of Provisional Operating License No. DPR-16 is hereby amended to read as follows:

   (B) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 24, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 19, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 24

TO THE TECHNICAL SPECIFICATIONS

PROVISIONAL OPERATING LICENSE NO. DPR-16

DOCKET NO. 50-219

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised page is identified by Amendment number and contains vertical lines indicating the area of change.

Remove	Replace
2.1-1	2.1-1
2.3-4	2.3-4
3.10-1 thru 3.10-4	3.10-1 thru 3.10.4
Figure 3.10.1	3.10-8 (Figure 3.10-1)

SECTION 2

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMIT - FUEL CLADDING INTEGRITY

Applicability: Applies to the interrelated variables associated with fuel thermal behavior.

Objective: To establish limits on the important thermal hydraulic variables to assure the integrity of the fuel cladding.

Specifications: A. When the reactor pressure is greater than 600 psia, the combination of reactor core flow and reactor thermal power to water shall not exceed the limit shown on Figure 2.1.1 for any fuel type.

A.1 Figure 2.1.1 applies directly when the total peaking factor is less than or equal to the following:

Fuel Type II

a.	Axial peak at core midplane or below of	2.80
b.	Axial peak above core midplane of	2.55

Fuel Types IIIE and IIIF

a.	Axial peak at core midplane or below of	2.74
b.	Axial peak above core midplane of	2.50

For 8 x 8 Fuel

a.	Axial peak at core midplane or below of	2.78
b.	Axial peak above core midplane of	2.61

A.2 For total peaking factors greater than those specified in Specification 2.1.A.1, the safety limit is reduced by the following:

$$SL = SL_o \times \frac{PFo}{PF}$$

where: SL = reduced safety limit
SL_o= safety limit from Figure 2.1.1
PF_o= peaking factor specified in Specification 2.1.A.1
PF = actual peaking factor

B. When the reactor pressure is less than 600 psia or reactor flow is less than 10 percent of design, the reactor thermal power shall not exceed 354 Mwt.

C. The neutron flux shall not exceed its scram setting for longer than 1.75 seconds.

evaluation of the reactor dynamic performance during normal operation as well as expected maneuvers and the various mechanical failures, it was concluded that sufficient protection is provided by the simple (3, 4). However, in response to expressed beliefs (5) that variation of APRM flux scram with recirculation flow is a prudent measure to ensure safe plant operation during the design confirmation phase of plant operation, the scram setting will be varied with recirculation flow. If during the power demonstration run the design analyses are confirmed with respect to nuclear behavior characteristics, the automatic flow biased scram could be replaced with a fixed scram setting.

Lowering the set point of the APRM scram would result in more margin between normal operation and the safety limit; however, lowering the set point could also result in spurious scrams. For example, there are transients which will occur during operation, such as those due to testing turbine bypass valves or pressure set point changes, which result in insignificant changes (<1%) in the power transferred from the cladding to the water, but for which the neutron flux rises 10-15%(3).

Calculations which include uncertainties in the heat balance show that the setting accuracy is $\pm$ 2.5% in the 85-100% power range (6). A turbine trip without bypass analyzed assuming a 125% scram showed no appreciable change in results from a 120% scram analysis (3). In addition, if the errors are random, some APRM's will trip low, the net effect being no change in the transient results. Therefore, allowing for instrument calibration errors, the scram setting is adequate to prevent the safety limit from being exeeeded and yet high enough to reduce the number of spurious scrams.

For slow power rises in the power range which might be produced by control rod withdrawal, the power is limited by the APRM control rod block(1), whose set point is varied automatically with recirculation flow. At conditions of rated flow or greater, the rod block is initiated at 106 percent of rated power. For the single rod withdrawal error this setting causes rod block before MCPR reaches 1.32 for 7 x 7 fuel and 1.34 for 8 x 8 fuel(13). For operation along the flow control line and at power levels less than 61% of rated the inadvertent withdrawal of a single control rod does not result in MCPR = 1.32 for 7 x 7 fuel and 1.34 for 8 x 8 fuel even assuming there is no control rod block action(7).

The safety curve of Figure 2.1.1 is based on total peaking factors of 2.74 for fuel types IIIE and IIIF: 2.80 for fuel type II; and 2.78 for 8 x 8 fuel. These curves are to be adjusted downward (by the equations shown in Specification 2.1.A.2) in the event of higher peaking factors. Also, to insure MCPR's greater than 1.32 for 7 x 7 fuel and 1.34 for 8 x 8 fuel during expected transients, neutron flux, scram and control rod block settings must be correspondingly reduced. The equations describing these setpoints make allowance for peaking factors greater than 2.74, 2.80, or 2.78 respectively for the fuel types listed above by reducing the setpoints at rated neutron flux by the ratio of PF_O to PF.

3.10 CORE LIMITS

Applicability: Applies to core conditions required to meet the Final Acceptance Criteria for Emergency Core Cooling Performance.

Objective: To assure conformance to the peak clad temperature limitations during a postulated loss-of-coolant accident as specified in 10 CFR 50.46 (January 4, 1974) and to assure conformance to the 17.2 KW/ft(for 7 x 7 fuel) and 14.5 KW/ft (for 8 x 8 fuel) operating limits for local linear heat generation rate.

Specification: A. Average Planar LHGR

During power operation, the average linear heat generation rate (LHGR) of all the rods in any fuel assembly, as a function of average planar exposure, at any axial location shall not exceed the maximum average planar LHGR shown in Figure 3.10-1. If at any time during power operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded, action shall be initiated to restore operation to within the prescribed limits. If the APLHGR is not returned to within the prescribed limits within two (2) hours, action shall be initiated to bring the reactor to the cold shutdown condition within 36 hours. During this period surveillance and corresponding action shall continue until reactor operation is within the prescribed limits at which time power operation may be continued.

B. Local LHGR

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

$$LHGR \leq LHGR_d \quad [1 - (\frac{\Delta P}{P}) \text{ max } \quad (\frac{L}{LT})]$$

Where: $LHGR_d$ = Limiting LHGR

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

LT = Total Core Length - 144 inches

L = Axial position above bottom of core

and

Fuel Type	$LHGR_d$	$\Delta P/P$
II	17.2	.032
IIIE	17.2	.046
IIIF	17.2	.033
V	14.5	.033
VB	14.5	.039

If at any time during operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded, action shall be initiated to restore operation to within the prescribed limits. If the LHGR is not retured to within the prescribed limits within two (2) hours, action shall be initiated to bring the reactor to the cold shutdown condition within 36 hours. During this period surveillance and corresponding action shall continue until reactor operation is within the prescribed limits at which time power operation may be continued.

C. Assembly Averaged Power Void Relationship

During power operation, the assembly average void fraction and assembly power shall be such that the following relationship is satisfied:

$$\left(\frac{1-VF}{PR \times FCP}\right) \geq B$$

Where: VF = Bundle average void fraction
PF = Assembly radial power factor
FCP = Fractional core power (relative to 1930 MWt)
B = Power-Void limit

The limiting values of "B" for each fuel type are shown in the table below:

Fuel Type(s)	B
II	.365
IIIE, IIIF	.377
V, VB	.332

D. During steady state power operation, MCPR shall be greater than or equal to the following:

ARPM Status	MCPR Limit
1. If any two (2) LPRM assemblies which are input to the APRM system and are separated in distance by less than three (3) times the control rod pitch contain a combination of three (3) out of four (4) detectors located in either the A and B or C and D levels which are failed or bypassed (i.e., APRM channel or LPRM input bypasses or inoperable).	1.64
2. If any LPRM input to the APRM system at the B, C, or D level is failed or bypasses or any APRM channel is inoperable (or bypassed).	1.58
3. All B, C and D LPRM inputs to the APRM system are operating and no APRM channels are inoperable or bypassed.	1.52

When ARRM status changes due to instrument failure (APRM or LPRM input failure), the MCPR requirement for the degraded condition shall be met within a time interval of eight (8) hours, providing that the control rod block is placed in operation during this interval.

If at any time during power operation it is determined by normal surveillance that the limiting value for MCPR is being exceeded for reasons other than instrument failure, action shall be initiated to restore operation to within the prescribed limits. If the steady state MCPR is not returned to within the prescribed limits within two (2) hours, action shall be initiated to bring the reactor to the cold shutdown condition within 36 hours. During this period surveillance and corresponding action shall continue until reactor operation is within the prescribed limits at which time power operation may be continued.

Basis: The Specification for average planar LHGR assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the 2200°F limit specified in 10 CFR 50.46 (January 4, 1974) considering the postulated effects of fuel pellet densification.

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is only dependent secondarily on the rod to rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than ± 20°F relative to the peak temperature for a typical fuel design, the limit on the average linear heat generation rate is sufficient to assure that calculated temperatures are below the limits specified in 10 CFR 50.46 (January 4, 1974).

The maximum average planar LHGR shown in Figure 3.10-1 for Type II fuel is the result of LOCA analyses performed utilizing a blowdown thermal-hydraulic analysis developed by General Electric Company in compliance with 10 CFR 50, Appendix K (January 4, 1974). Single failure considerations were based on the revised Oyster Creek Single Failure Analysis submitted to the Staff on July 15, 1975.

The maximum average planar LHGR shown in Figure 3.10-1 for Type IIIE, IIIF, V and VB fuel are the result of Appendix K approved LOCA analyses performed by Exxon Nuclear Company utilizing blowdown results obtained from General Electric Company which reflect revised single failure considerations.(1) In addition, the maximum average planar LHGR shown in Figure 3.10-1 for Type V and VB fuel were analyzed with 100% of the spray cooling coefficients specified in Appendix K to 10 CFR Part 50 for 7x7 fuel. These spray heat transfer coefficients were justified in the END Spray Cooling Heat Transfer Test Program.(2)(3).

The possible effects of fuel pellet densification are: 1) creep collapse of the cladding due to axial gap formation; 2) increase in the LHGR because of pellet column shortening; 3) power spikes due to axial gap formation; and 4) changes in stored energy due to increased radial gap size.

Calculations show that clad collapse is conservatively predicted not to occur during the exposure lifetime of the fuel. Therefore, clad collapse is not considered in the analyses. Since axial thermal expansion of the fuel pellets is greater than axial shrinkage due to densification, the analyses of peak clad temperature do not consider any change in LHGR due to pellet column shortening. Although the formation of axial gaps might produce a local power spike at one location on any one rod in a fuel assembly, the increase in local power density would be on the order of only 2% at the axial midplane. Since small local variations in power distribution have a small effect on peak clad temperature, power spikes were not considered in the analysis of loss-of-coolant accidents.

Changes in gap size affect the peak clad temperatures by their effect on pellet clad thermal conductance and fuel pellet stored energy. Treatment of this effect combined with the effects of pellet cracking, relocation and subsequenc gap closure are discussed in NEDO-20181 and XN-174.

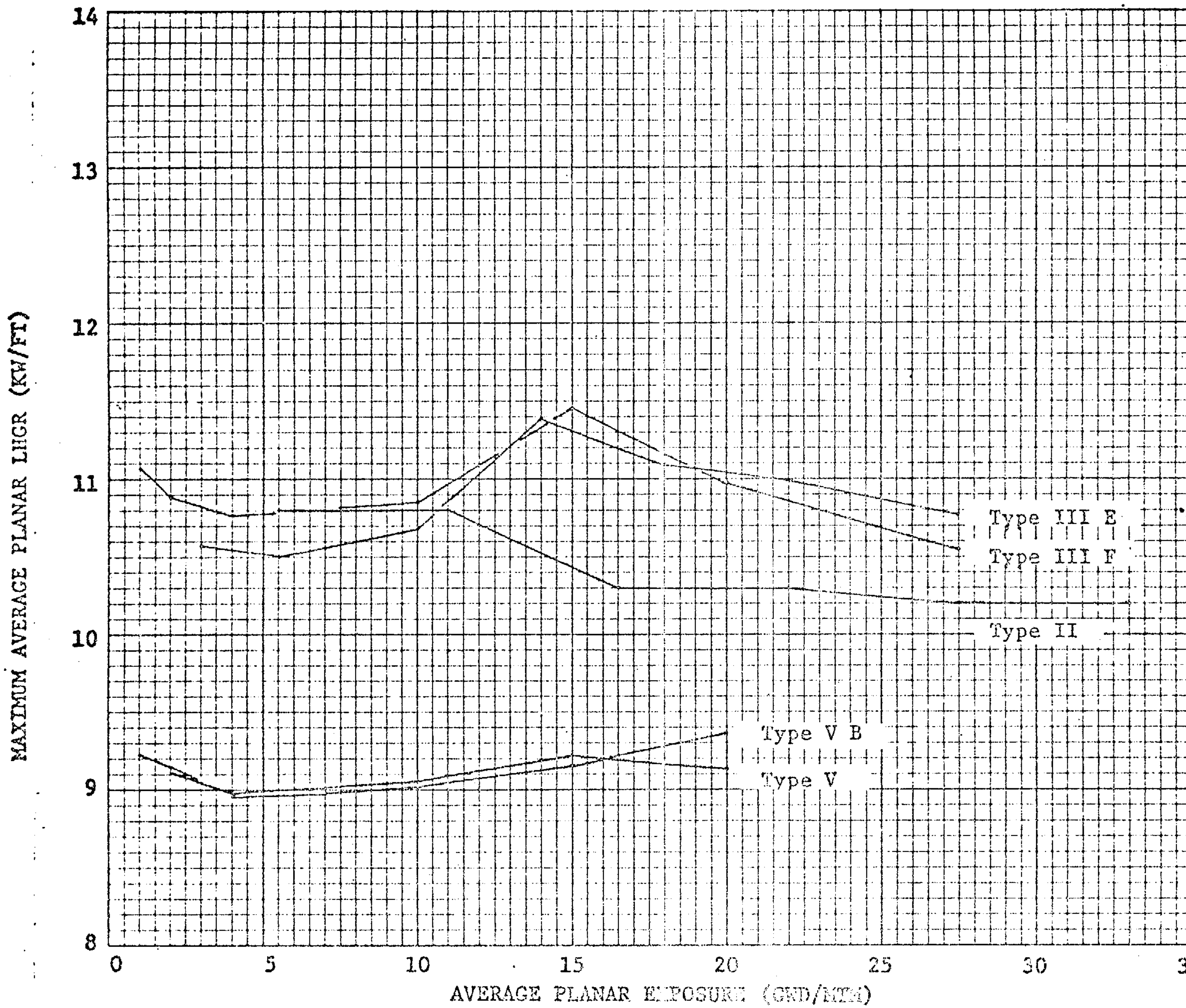
Pellet-clad thermal conductance for Type II fuel was calculated using the GEGAP III model (NEDO-20181) and Pellet-clad thermal conductance for Type IIIE, IIIF, V and VB fuel was calculated using the GEPEX model (XN-174).

The specification for local LHGR assures that the linear heat generation rate in any rod is less than the limiting linear heat generation even if fuel pellet densification is postulated. The power spike penalty specified for Type II fuel is based on the analysis presented in Section 3.2.1 of the GE Topical Report NEDM-10735 Supplement 6. The power spike penalty for Type IIIE, and IIIF fuel is based on analyses presented in Facility Change Request Nos. 4 and 5, Facility Change Request No. 6 for Type V and Amendment No. 76 for Type VB fuel. The analysis assumes a linearly increasing variation in axial gaps between core bottom and top, and assures with 95% confidence that no more than one fuel rod exceeds the design linear heat generation rate due to power spiking.

The specification on the assembly averaged power-void relationship provides assurance that operating conditions will be more conservative than the initial conditions assumed in the LOCA analysis, therefore assuring applicability of the analyses.

FIGURE 3.10.1

MAXIMUM ALLOWABLE AVERAGE PLANAR
LINEAR HEAT GENERATION RATE



Amendment No. ~~16~~, 24

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 24 TO LICENSE NO. DPR-16

JERSEY CENTRAL POWER AND LIGHT COMPANY

OYSTER CREEK NUCLEAR POWER PLANT UNIT NO. 1

DOCKET NO. 50-219

Introduction

By letter dated June 23, 1977, Jersey Central Power and Light Company (JCP&L) requested an amendment to Facility Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station. The amendment would modify the Technical Specifications to delete specifications and bases applicable to fuel types I and III which are no longer in the Oyster Creek core and to extend the MAPLHGR curves for the fuel types still in the core.

Discussion and Evaluation

JCP&L had removed all type I and type III fuel during the spring outage that started April 23, 1977. The present core consists of an array of fuel types II, IIIE, IIIF, V and VB. In the application for amendment, JCP&L has identified all of the areas in the Technical Specifications and bases where reference is made to fuel types I and III. Since deletion of these references only involve an administrative change we conclude that the proposed changes are acceptable.

In the application for amendment, JCP&L has extended the Maximum Allowable Average Plannar Linear Heat Generation Rate specification for fuel types currently in the core by modifying Figure 3.10.1. This modification extends the calculated Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) from 15 GWD/MTM (thousand megawatt days/metric ton) to 20 GWD/MTM for fuel types V and VB, from 20 GWD/MTM to 27.5 GWD/MTM for fuel type IIIF and from 22 GWD/MTM to 27.5 GWD/MTM for fuel type IIIE. The calculated MAPLHGR for fuel type II was not extended.

The extensions of MAPLHGR shown in Figure 3.10-1 for types IIIE, IIIF, V, and VB fuel are the results of an analysis using an approved ECCS evaluation model in agreement with Appendix K to 10 CFR Part 50. This LOCA analyses, performed by Exxon Nuclear Company (ENC), utilized blowdown results obtained from General Electric Company which reflect revised single failure considerations. In addition, the MAPLHGR extensions shown in Figure 3.10-1 for types V and VB fuel were analyzed with 100% of the spray cooling coefficient specified in Appendix K to 10 CFR Part 50 for 7 x 7 fuel.

Use of the Appendix K LOCA analyses performed by ENC and JCP&L was justified because an earlier license amendment request, based on reevaluation of the Oyster Creek ECCS, was revised and found acceptable by the NRC staff. In the later licensing action, the staff issued License Amendment No. 15 dated February 24, 1976, to Provisional Operating License (POL) No. DPR-16 which documents our findings on the ECCS Evaluation model and its application to Oyster Creek.

Use of 100% spray heat transfer coefficients specified in Appendix K was justified in the Exxon Nuclear Company (ENC) Spray Cooling Heat Transfer Test Program. As a result the NRC staff issued Licnense Amendment No. 16 to POL No. DPR-16 dated July 26, 1976 which documents our findings on the review and NRC staff acceptance of this application of heat transfer coefficient to Oyster Creek.

We have reviewed the proposed changes and have concluded that they do use the methods found acceptable by the NRC staff, do not change MAPLHGR limits previously found acceptable and only extend the current calculations to higher fuel exposures and therefore are acceptable.

Environmental Considerations

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR §51.5(d)(4) that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed aobve, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards considerations, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: September 19, 1977

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-219

JERSEY CENTRAL POWER & LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENT

TO PROVISIONAL OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 24 to Provisional Operating License No. DPR-16 issued to Jersey Central Power & Light Company which revised Technical Specifications for operation of the Oyster Creek Nuclear Generating Station, located in Ocean County, New Jersey. The amendment is effective as of its date of issuance.

The amendment modifies the Technical Specifications to delete specifications and bases applicable to fuel types I and III which are no longer in the Oyster Creek core and to extend the MAPLHGR curves for the fuel types still in the core.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated June 23, 1977, (2) Amendment No. 24 to License No. DPR-16, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Ocean County Library, Brick Township Branch, 401 Chambers Bridge Road, Brick Town, New Jersey 08723. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 19 day of September 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors