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. 8 to Provisional Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station, Unit 1. The amendment includes Change No. 24 to the Technical Specifications and is in response to your applications dated March 25 and 29, 1975, and April 30, 1975.

The amendment incorporates operating limits in the Technical Specifications for the facility based on an acceptable evaluation model that conforms with the requirements of Section 50.46 of 10 CFR Part 50. A copy of the related Safety Evaluation is enclosed.

The Commission's staff has evaluated the potential for environmental impact associated with operation of Oyster Creek Nuclear Generating Station, Unit 1 in the manner proposed. From this evaluation, the staff has determined that there will be no change in effluent types or total amounts, no increase in authorized power level and no significant environmental impact attributable to the proposed action. Having made this determination, the Commission has further concluded, pursuant to 10 CFR Part 51, Section 51.5(c)(1) that no environmental impact statement need be prepared for this action. Copies of the related Negative Declaration and supporting Environmental Impact Appraisal are enclosed. As required by Part 51, the Negative Declaration is being filed with the Office of the Federal Register for publication.

Your attention is directed to the fact that a restriction has been incorporated in the license that does not permit operation with one or more recirculation loops out of service. This restriction was incorporated because information was not provided for our evaluation of ECCS performance during reactor operation with recirculation loops out of service.

The license also contains conditions requiring submittal of proposed design modifications to enable the facility to automatically accommodate a single passive failure of the diesel generator bus without adverse effect on the ECCS performance. The proposed modifications are to be submitted for approval within 30 days and should be installed within 30 days after they are approved by the NRC. You are also required to submit within 30 days, your complete assessment verifying that your facility will accommodate single passive electrical failures without adverse effect on ECCS performance.

In addition to documents described above, a copy of the related Federal Register Notice is enclosed.

Sincerely,

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

Enclosures:
1. Amendment No. 8
2. Safety Evaluation
3. Negative Declaration
4. Environmental Impact Appraisal
5. Federal Register Notice

cc: w/enclosures
See next page

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cc w/encls:
G. F. Trowbridge, Esquire
Shaw, Pittman, Potts, Trowbridge & Madden
Barr Building
910 17th Street, N. W.
Washington, D. C. 20006

Jersey Central Power & Light Company
ATTN: Mr. Thomas M. Crimmins, Jr.
Safety and Licensing Manager
GPU Service Corporation
260 Cherry Hill Road
Parsippany, New Jersey 07054

Anthony Z. Roisman, Esquire
Berlin, Roisman and Kessler
1712 N Street, N. W.
Washington, D. C. 20036

Paul Rosenberg, Esquire
Daniel Rappoport, Esquire
2323 S. Broad Street
Trenton, New Jersey 08610

Honoable Joseph W. Ferraro, Jr.
Deputy Attorney General
State of New Jersey
101 Commerce Street - Room 208
Newark, New Jersey 07102

Burtis W. Horner, Esquire
Stryker, Tams and Dill
55 Madison Avenue
Morristown, New Jersey 07960

George F. Kugler, Jr.
Attorney General
State of New Jersey
State House Annex
Trenton, New Jersey 08625

The Honorable W. M. Mason
Mayor, Lacey Township
P. O. Box 475
Forked River, New Jersey 08731

cc w/encls and JCP&L's filings
dtd. 4/25/75 and 4/29/75
Honorable William F. Hyland
Attorney General
State of New Jersey
State House Annex
Trenton, New Jersey 08601

Mr. Paul Arbesman
Environmental Protection Agency
Region II Office
26 Federal Plaza
New York, New York 10007

OFFICE ➤						
SURNAME ➤						
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

JERSEY CENTRAL POWER & LIGHT COMPANY

DOCKET NO. 50-219

(OYSTER CREEK NUCLEAR GENERATING STATION)

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 8
License No. DPR-16

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

   A. The applications for amendment by Jersey Central Power & Light Company (the licensee) dated March 25 and 29, 1975, comply 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 applications, 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; and

   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.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and Paragraphs 3.B and 3.C of Provisional Operating License No. DPR-16 are amended and added (respectively) to read as follows:

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"B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 24."

"C. Recirculation Loop Inoperable

The reactor shall not be operated with one or more recirculation loops out of service."

"D. Electrical Supply for ECCS System

1. Within 30 days from the effective date of this amendment the licensee shall submit for NRC review and approval proposed design modifications which will enable the facility to automatically accommodate a single passive failure of the emergency diesel generator bus without adverse effect on the ability of the ECCS system to conform to the evaluation submitted to demonstrate compliance with 10 CFR 50.46 in the applications for license amendment set forth above. Such modifications shall be completed within 30 days after approval, or within such other time as may be specified in such approval.

2. Within 30 days from the effective date of this license amendment, the licensee shall submit a complete re-assessment of all elements of the electrical systems associated with ECCS performance to verify that no single passive electrical failure would adversely affect the ability of the ECCS to conform to the evaluation submitted to demonstrate compliance with 10 CFR 50.46. If any such failure is identified prior to the end of such 30 day period, the NRC shall be informed promptly upon identification of such a potential failure.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Karl R. Goller /for

A. Giambusso, Director
Division of Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Change No. 24 to the
Techncial Specifications

Date of Issuance: MAY 2 4 1975

ATTACHMENT TO LICENSE AMENDMENT NO. 8

CHANGE NO. 24 TO THE TECHNICAL SPECIFICATIONS

PROVISIONAL OPERATING LICENSE NO. DPR-16

DOCKET NO. 50-219

Replace pages 3.10-1 through 3.10-3 and 4.10-1 with the attached revised pages.

Replace Figure 3.10.1 with the attached revised figure.

3.10 ECCS RELATED CORE LIMITS

Applicability: Applies to core conditions required to meet the Final Acceptance Criteria for Emergency Core Cooling Performance. 24

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 steady state 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.

B. Local LHGR

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

$$LHGR \leq LHGR_d \left[1 - \left(\frac{\Delta P}{P}\right)_{max} \cdot \left(\frac{L}{LT}\right)\right]$$ 24

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	$LHGR_d$	$\Delta P/P$
I	17.2	.038
II	17.2	.032
III	17.2	.046
IIIE	17.2	.046
IIIF	17.2	.033
V	14.5	.033
VB	14.5	.039

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:

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

Where: VF = Bundle average void fraction
PF = Assembly radial power factor 24
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
I, II, III	.365
IIIE, IIIF	.377
V, VB	.332

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 I and II fuel are 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) 24
and submitted to the Staff on March 29, 1975 as required in the Staff "Order for Modification of License for the Oyster Creek Nuclear Generating Station", dated December 27, 1974.

The maximum average planar LHGR shown in Figure 3.10.1 for Type III, IIIE, IIIF, V and VB fuel are the result of LOCA analyses performed by Exxon Nuclear Company utilizing blowdown results obtained from General Electric Company and submitted to the Staff on April 30, 1975 as required in the Staff "Order for Modification of License for the Oyster Creek Nuclear Generating Station", dated December 27, 1974.

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, 24
clad collapse is not considered in the analyses. Since axial
thermal expansion of the fuel pellets is greater than axial shrink-
age 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 in-
crease in local power density would be on the order of only 2% at
the axial midplane. Since small local variations in power distri-
bution 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 temperature 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 subsequent gap closure are discussed
in NEDO-20181 and XN-174.

Pellet-clad thermal conductance for Type I and II fuel was calcu-
lated using the GEGAP III model (NEDO-20181) and Pellet-clad
thermal conductance for Type III, IIIE, IIIF, V and VB fuel was 24
calculated using the GAPEX 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 I and 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
Types III, 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 24
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 conserva-
tive than the initial conditions assumed in the LOCA analysis, there-
fore assuring applicability of the analyses. Non-jet pump BWR ECCS
models utilize an empirical correlation to determine the duration 24
of nucleate boiling heat transfer in the early period following
the postulated pipe break. This correlation for time to dryout
is found to be proportional to the ratio of assembly water volume
to power. Dryout time is a significant parameter in determining
the extent of nucleate and transition boiling heat transfer, and
consequently the peak cladding temperature.

By maintaining reactor power and void fraction as specified in
3.10.C, dryout times at least as long as that used in the LOCA
analysis will be assured. The limiting values of B shown in the

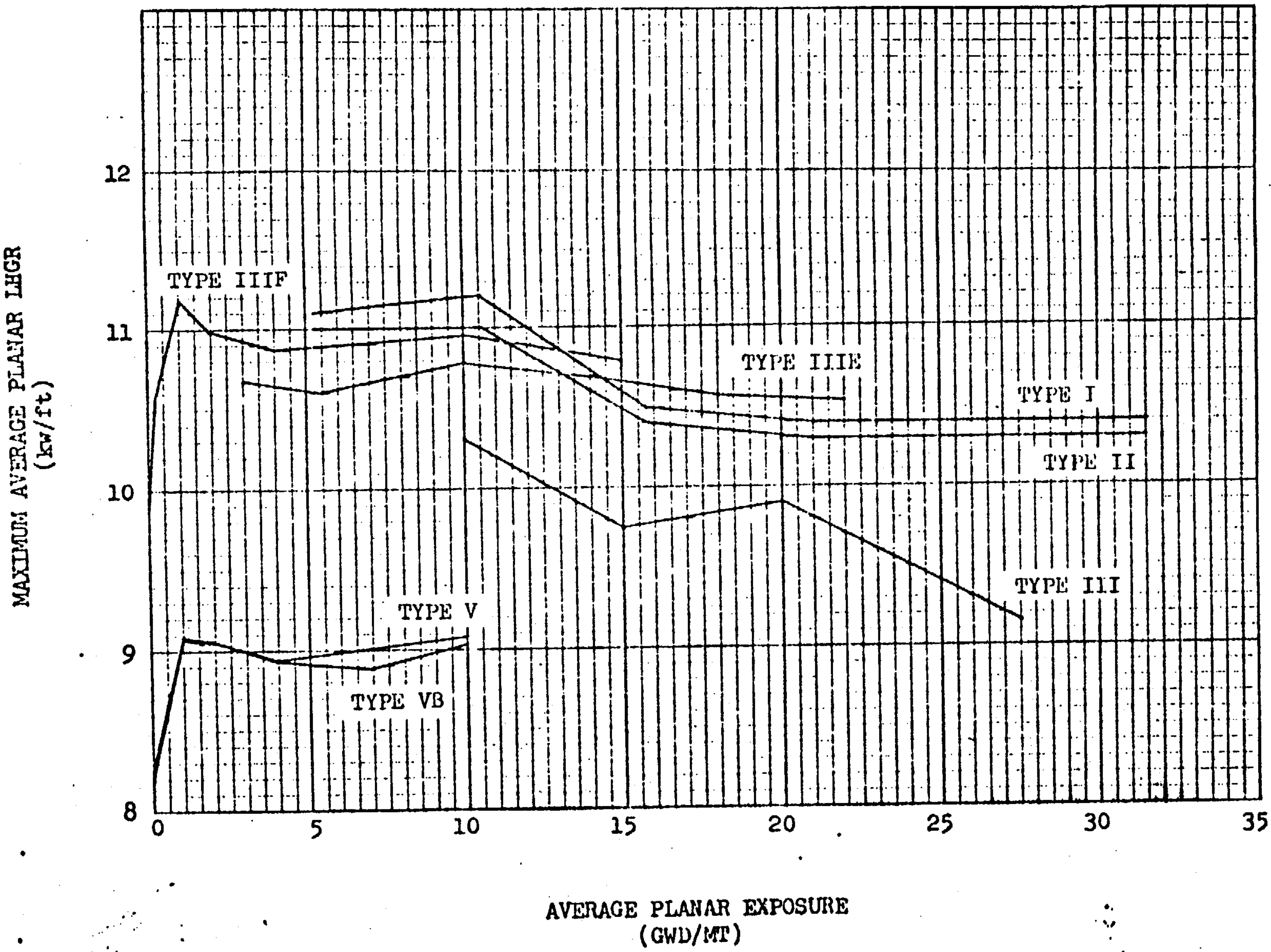
table in Specification 3.10.C above were developed for core conditions of 100% power and 70% flow, the minimum flow that could be achieved without automatic plant trip (flow biased high neutron flux scram). Such a condition is never achieved during actual operation due to the neutron flux rod block and the inherent reactor power-flow relationship. The MAPLHGR results shown in Figure 3.10.1 were evaluated for 102% power and 70% flow, thus the 2% conservatism for instrument uncertainty is retained in the limiting values of B shown in the table. Additional conservatism is provided by the following assumptions used in determining the B limits: 24

1. All heat was assumed to be removed by the active channel flow. No credit was taken for heat removal by leakage flow (10% of total flow).

2. Each fuel type was assumed to be operating at full thermal power rather than the reduced power resulting from the more limiting conditions imposed by Figure 3.10.1.

FIGURE 3.10.1

MAXIMUM ALLOWABLE AVERAGE PLANAR
LINEAR HEAT GENERATION RATE



4.10 ECCS RELATED CORE LIMITS

Applicability: Applies to the periodic measurement during power operation of core parameters related to ECCS performance. 24

Objective: To assure that the limits of Section 3.10 are not being violated.

Specification:

A. Average Planar LHGR

Daily during reactor power operation, the average planar LHGR shall be checked.

B. Local LHGR

Daily during reactor power operation, the local LHGR shall be checked.

C. Assembly Averaged Power-Void Relationship

Compliance with the Power-Void Relationship in Section 3.10.C will be verified at least once during a startup between 50% and 70% power, when
steady state power operation is attained, and at 24
least every 72 hours thereafter during power operation.

Basis:

The LHGR shall be checked daily to determine whether fuel burnup or control rod movement has caused changes in power distribution. Since changes due to burnup are slow, and only a few control rods are moved daily, a daily check of power distribution is adequate.

The Power-Void Relationship is verified between 50% and 70% power during a startup. This single verification during startup is acceptable since operating experience has shown that even under the most extreme void conditions encountered at lower power levels, the relationship is not violated.
Additionally, reduced power operation involves less stored 24
heat in the core and lower decay heat rates which would add further margin to limiting peak clad temperatures in the event of a LOCA.

Verification when steady state power operation is attained and every 72 hours thereafter is appropriate since once steady state conditions are achieved, the void fraction, radial peaking factor, and power level that combine to form the relationship are unlikely to change so rapidly to result in a significant change during that period.

NEGATIVE DECLARATION

REGARDING PROPOSED CHANGES TO THE

TECHNICAL SPECIFICATIONS OF LICENSE DPR-16

OYSTER CREEK NUCLEAR GENERATING STATION

DOCKET NO. 50-219

The Nuclear Regulatory Commission (the Commission) has considered the issuance of changes to the Technical Specifications of Provisional Operating License No. DPR-16. These changes would authorize the Jersey Central Power & Light Company (the licensee) to operate the Oyster Creek Nuclear Generating Station (located 10 miles south of Toms River, New Jersey) with changes to the limiting conditions for operation associated with fuel assembly specific power (average planar linear heat generation rate) resulting from application of the Acceptance Criteria for Emergency Core Cooling System (ECCS). This change is being made in conjunction with a partial core refueling with 8 x 8 fuel.

The U. S. Nuclear Regulatory Commission, Division of Reactor Licensing, has prepared an environmental impact appraisal for the proposed changes to the Technical Specifications of License No. DPR-16, Oyster Creek Nuclear Generating Station, described above. On the basis of this appraisal, the Commission has concluded that an environmental impact statement for this particular action is not warranted because there will be no environmental impact attributable to the proposed action other than that which has already been predicted and described in the Commission's Final Environmental Statement for Oyster Creek Nuclear Generating Station issued in December 1974.

The environmental impact appraisal is 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, 5 Hooper Avenue, Toms River, New Jersey.

Dated at Rockville, Maryland, this 30th day of April, 1975.

FOR THE NUCLEAR REGULATORY COMMISSION

Wm. H. Regan

Wm. H. Regan, Jr., Chief
Environmental Projects Branch 4
Division of Reactor Licensing

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

ENVIRONMENTAL IMPACT APPRAISAL BY THE DIVISION OF REACTOR LICENSING

SUPPORTING AMENDMENT NO. 8 TO DPR-16

CHANGE NO. 24 TO THE TECHNICAL SPECIFICATIONS

JERSEY CENTRAL POWER & LIGHT COMPANY

OYSTER CREEK NUCLEAR GENERATING STATION

ENVIRONMENTAL IMPACT APPRAISAL

1. Description of Proposed Action

By letters dated March 29, 1975 and April 30, 1975 the Jersey Central Power and Light Company submitted proposed changes to the Technical Specifications Appendix A to License No. DPR-16. The proposed changes were requested to incorporate limiting conditions for operation associated with fuel assembly specific power (average planar linear heat generation rate) resulting from the application of the Acceptance Criteria for Emergency Core Cooling System (ECCS) in conjunction with a partial core refueling using 8 x 8 fuel. The staff has reviewed this matter and the conclusions are set forth below.

The licensee is presently licensed to possess and operate Oyster Creek Nuclear Generating Station located in the State of New Jersey, County of Ocean, at power levels up to 1,930 megawatts thermal (MWt). The proposed change to incorporate the ECCS Acceptance Criteria in conjunction with a partial core refueling using 8 x 8 fuel may result in a decrease in total electrical plant generation of the unit by an estimated 10 to 16%. The reduced electrical generation is the result of conservative assumptions that the licensee made in the ECCS Evaluation Model in order to reduce the complexity of the model. When the licensee develops more realistic thermal hydraulic models for use in his ECCS evaluation, the reactor can subsequently be authorized to operate at or near full power. Since the power level is not significantly affected by the action, the action does not affect the benefits of electric power production considered for the captioned facility in the Commission's Final Environmental Statement (FES) for Oyster Creek Nuclear Generating Station, Docket No. 50-219 dated December 1974.

2. Environmental Impacts of Proposed Action

Potential environmental impacts associated with the proposed action are those which may be associated with incorporation of the ECCS Acceptance Criteria and utilization of nuclear fuel for this facility.

It is particularly noted that in the absence of any significant change in power levels, there will be no significant change in cooling water requirements and consequently no significant increase in environmental impact from radioactive effluents and thermal effluents for normal operation or post-accident conditions which in turn could not lead to significant increases in radiation doses or thermal stress to the public or to biota in the environment.

For normal operating conditions, no environmental impact other than as described in the Commission's Final Environmental Statement (FES) for Oyster Creek Nuclear Generating Station, Docket No. 50-219 dated December 1974, can be predicted for the proposed action. The Commission's calculated releases for radioactive effluents, both gaseous and liquid, are based on expected release rates to the environment and are quantified on the basis of the total quantity of nuclear fuel within the reactor. The estimates of radionuclide releases will not be affected by the proposed action, and since the total quantity of nuclear fuel is unchanged, no increase in the calculated release of radioactive effluents is predicted. Consequently, no significant increases in radiation doses to man or other biota are predicted.

3. Conclusion and Basis for Negative Declaration

On the basis of the foregoing analysis, it is concluded that there will be no environmental impact attributable to the proposed action other than has already been predicted and described in the Commission's FES for Oyster Creek Nuclear Generating Station. Having made this conclusion, the Commission has further concluded that no environmental impact statement for the proposed action need be prepared and that a negative declaration to this effect is appropriate.

MAY 2 4 1975

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 8 TO PROVISIONAL OPERATING LICENSE NO. DPR-16

(CHANGE NO. 24 TO THE TECHNICAL SPECIFICATIONS)

JERSEY CENTRAL POWER & LIGHT COMPANY

OYSTER CREEK NUCLEAR GENERATING STATION

Introduction

On December 27, 1974, the Atomic Energy Commission issued an Order for Modification of License implementing the requirements of 10 CFR § 50.46 "Acceptance Criteria and Emergency Core Cooling Systems for Light Water Nuclear Power Reactors." One of the requirements of the Order was that prior to any license amendment authorizing any core reloading" ... the licensee shall submit a reevaluation of ECCS cooling performance calculated in accordance with an acceptable evaluation model which conforms with the provisions of 10 CFR Part 50, § 50.46." The Order also required that the evaluation shall be accompanied by such proposed changes in Technical Specifications or license amendments as may be necessary to implement the evaluation results.

By letter dated March 25, 1975, Jersey Central Power & Light Company (JCP&L) submitted a description of the JCP&L ECCS Evaluation Model to be used to perform the ECCS reevaluation for the Oyster Creek Nuclear Generating Station. The JCP&L Evaluation Model uses the General Electric Company (GE) Evaluation Model to perform the ECCS reevaluation for GE fuel and couples the GE Evaluation Model and Exxon Nuclear Company (Exxon) Fuel Heatup Model to perform the ECCS reevaluation for the Exxon fuel. On March 29, 1975, JCP&L submitted an evaluation of ECCS performance for the design basis piping break for the Oyster Creek Nuclear Generating Station and requested an amendment to Provisional Operating License No. DPR-16 to change the Technical Specifications for Oyster Creek Nuclear Generating Station to implement the results of the evaluation. By letter dated April 4, 1975, JCP&L provided additional details of the ECCS evaluation. By letters dated April 24, April 28 (including proprietary figures), April 30, 1975, and May 5, 1975, JCP&L provided further information relating to the details of the ECCS analysis requested by us. By letter dated April 30, 1975, JCP&L provided additional information requested by us and submitted the complete

AMERICAN REVOLUTION BICENTENNIAL 1776-1976

ECCS reevaluation for the Exxon fuel which superseded the information regarding the Exxon fuel in the March 29, 1975 submittal. The JCP&L evaluation of ECCS performance was based on the reactor with a partial reload of 8 x 8 fuel. Our review of the use of 8 x 8 fuel in the Oyster Creek Nuclear Generating Station is the subject of a separate safety evaluation. Notice of the proposed action regarding the ECCS reevaluation was published in the Federal Register on April 4, 1975 (40 FR 15137).

Evaluation

1. ECCS Evaluation Model

The Order for Modification of License issued December 27, 1974, stated that evaluation of ECCS cooling performance may be based on the General Electric evaluation model as modified in accordance with the changes described in our Safety Evaluation Report of the Oyster Creek Nuclear Generating Station dated December 27, 1974, and on an acceptable Exxon model, as corrected.

a. General Electric Company's Evaluation Model

The background of the Nuclear Regulatory Commission (NRC) staff review of the General Electric (GE) ECCS Evaluation Model and the application to the Oyster Creek Nuclear Generating Station is described in our Safety Evaluation Report (SER) issued in connection with the Order dated December 27, 1974. The bases for acceptance of the principal portions of the GE Evaluation Model are set forth in our Status Report of October 1974 and the Supplement to the Status Report of November 1974. The December 27, 1974 SER also describes the various changes required in the earlier GE Evaluation Model. Together, the December 27, 1974 SER and the Status Report and its Supplement describe an acceptable GE ECCS Evaluation Model and the basis for our acceptance of their model. Therefore, those portions of the Jersey Central Power and Light Company ECCS Evaluation which utilize the GE ECCS Evaluation Model properly conform to the requirements of Appendix K to 10 CFR Part 50.

b. Exxon's Fuel Heatup Model

The Exxon model performs fuel heatup calculations for Exxon fuel using, as input, certain parameters dependent primarily on the design and function of the Nuclear Steam Supply System (NSSS). Those parameters are taken from intermediate results available from the GE model referenced above. Our review of the Exxon model is described in Reporting Regarding the Exxon Nuclear Company ECCS Non-Jet-Pump-BWR Fuel Heatup Model (NJP-BWR-FHM), March 6, 1975. That report describes an Exxon NJP-BWR-FHM which is acceptable for use as part of an applicant's ECCS Evaluation Model when properly coupled with the GE ECCS Evaluation Model referenced above. Those portions of the Jersey Central Power and Light Company ECCS Evaluation which utilize the Exxon NJP-BWR-FHM properly conform to the requirements of Appendix K to 10 CFR Part 50.

c. Jersey Central Power & Light Company's Evaluation Model

The Jersey Central Power and Light Company ECCS Evaluation Model, which is a composite of the GE and Exxon models, is described in their letter of March 25, 1975. The JCP&L model consists of (1) the GE ECCS Evaluation Model for analyses of GE fuel, and (2) a proper combination of the GE ECCS Evaluation Model and the Exxon NJP-BWR-FHM for analyses of Exxon fuel. Based upon our review of the March 25, 1975 letter describing the JCP&L ECCS Evaluation Model, we conclude that the model as described therein and as used for the ECCS analyses conforms to all requirements of Appendix K to 10 CFR Part 50 and that it is appropriately applied to the Oyster Creek reactor.

d. Plant Specific Items

Our generic Report Regarding the Exxon ECCS-NJP-BWR-FHM (March 6, 1975) specified certain items that must be provided or justified on a case by case basis. The following subparagraphs discuss the fulfillment, by the licensee and Exxon, of these requirements:

(1) Conservative Application of GAPEX

It was to be demonstrated on a case by case basis that volumetric average fuel temperature for each node in the HUXY calculation was greater than or equal to that value as calculated by GAPEX, in order to ensure that the GAPEX information was being used in a conservative manner by the heatup (HUXY) calculation. Although stated as a "case by case" item in the generic report, the example calculations provided during the generic model review were developed for Oyster Creek; hence, the present application of the model to Oyster Creek has already been demonstrated in an acceptable manner.

(2) Location of Axial Plane in Which Peak Clad Temperatures Occur

It was to be demonstrated on a case by case basis that the plane of interest (POI) for which peak clad temperatures are calculated is in fact the axial plane in which the maximum temperatures would occur. Although stated in the generic report as a "case by case" item, the example calculations provided during the generic model review were developed for Oyster Creek; hence, the present application to Oyster Creek has already been demonstrated in an acceptable manner.

(3) Rod Bowing

The generic report stated "In each individual plant submittal employing the Exxon model, the applicant will be required to properly take rod bowing into account." Based on similarity of Exxon and GE fuel and on results of a GE study which showed that postulated fuel rod bowing does not increase the maximum peak clad temperature which occurs during the LOCA event, the licensee concludes that the clad temperature is not increased because (1) rod bowing occurs in a plane of lower power than the "peak plane" (swelling of the fuel rods located in the peak power plant is the postulated cause of the bowing), and (2) the lower power in the plane in which the rods are bowed together more than compensates for the lower heat transfer that results from the bowed rod geometry. We find this explanation of the effect of rod bowing to be acceptable.

(4) Adequacy of NSSS Vendor Information

The generic report states that "Adequacy of detail of information available from the NSSS vendor will be judged on a case by case basis." We reviewed JCP&L's ECCS Evaluation Model and calculations, including input data made available from the NSSS vendor for use with the Exxon NJP-BWR-FHM. We conclude that the degree of detail present in the NSSS vendor-supplied information is adequate and, therefore, is acceptable. Moreover, we have assured by our review that the results contain no unacceptable uncertainties due to lack of precision or detail in the NSSS vendor-supplied information.

(5) Fraction of Locally Generated Gamma Energy Deposited in the Fuel Pin

The generic report states "The fraction of locally generated gamma energy deposited in the fuel pin may be less than 1.0 if justified by calculations." A value of 0.967 (3.33% deposited externally to the fuel pin in which it is generated] is used in the Exxon fuel analyses for the Oyster Creek Nuclear Generating Station. JCP&L presented an evaluation which was based upon the similarity of Exxon and GE fuel and which also referred to certain GE proprietary data that properly shows the minimum value exceeds the 3.33% external deposition of gamma energy employed by Exxon in the present analysis. We concur that 3.33% external energy deposition is a conservative value and is therefore acceptable for use in these analyses.

(6). Fission Power Curve

The generic report states "For small and intermediate size breaks, the applicability of the fission power curve used in the calculations will be justified by Exxon on a case by case basis. This will include justification of time of scram (beginning point in time of the fission power decrease) and the rate of fission power decrease due to voiding, if any." The licensee uses the Design Basis Accident (DBA) fission heat decay curve for all break sizes starting at the time of break with water level, at time of break, assumed to be at the low level scram point.

The licensee has provided a sensitivity study that shows that use of the DBA curve is justified for small and intermediate break analyses. The study demonstrated that there are insignificant effects on the final peak clad temperature (less than +10^{o}F) when credit is taken in the small and intermediate break analyses for power decreases due to voiding effects that are present in greater magnitude in the DBA than in the smaller breaks.

The assumption that the water level is at the low level scram point conservatively predicts the system inventory at the time shutdown starts. This method, however, does not address the power generation or heat transfer that might occur between the time of the break and the time the scram would actually occur if the water level was above the assumed low level scram point at time of break. This condition would be of concern only if a significant core flow decrease were to occur before the scram. Core flow decreases could be caused either by the flow from the break itself or by a recirculation pump trip due to assumed loss of offsite power. In the former case, if break flow were large enough to significantly decrease core flow, a scram would occur on high drywell pressure before occurrence of the assumed low level scram; and in the latter case, loss of offsite power would cause an even earlier scram.

For these reasons, we conclude that JCP&L's method of calculating fission power is acceptable.

2. Evaluation of ECCS Performance

The results presented in JCP&L's letters of March 29, 1975 and April 30, 1975 show the limiting break size to be a break area of 0.35 ft^2. This departure from the limiting break area of 4.69 ft^2 for the design basis break used in all previous analyses results from the lack of an approved analytical technique conforming to the new ECCS criteria (Appendix K to 10 CFR Part 50) for calculating flow coastdown in non-jet pump BWR plants such as Oyster Creek, following a LOCA. Therefore, credit for extended nucleate boiling of reactor coolant, following postulated small breaks, could not be assumed; as a result, the small break analyses produces a calculation of the same degraded heat transfer following a LOCA as does the large break analyses. In addition, a delay of approximately 162 seconds in onset of spray cooling results from use of a 0.35 ft^2 instead of the 4.69 ft^2 break size; this delay is due to the slower depressurization of the reactor vessel for the small breaks. Since the delay in onset of spray cooling causes longer adiabatic heating and subsequent rod heat-up, this causes the smaller break analyses to become accident condition. In other words, the breaksize of 0.35 ft^2 is the particular limiting small break because of its relation to the worst combination of hot plane uncovery time and the time between that uncovery and start of spray cooling (adiabatic heating time). Larger breaks uncover the hot plane earlier when decay powers are higher, but the quicker reactor vessel depressurization allows earlier core spray initiation with resulting shorter uncovery periods, a compensatory effect. Smaller breaks uncover the hot spot later when decay power is lower but have a longer adiabatic heating period due to slower depressurization and later spray initiation. The result of these two competing effects is that the calculated clad temperature peak occurs for a break area of 0.35 ft^2.

We have reviewed these results and conclude that, while they may be unrealistic due to lack of consideration of extended nucleate boiling for smaller breaks, they are conservative. These analyses using the 0.35 ft^2 break as limiting met all requirements of Appendix K to 10 CFR Part 50 and are acceptable.

During the course of our review, we concluded that additional recirculation line break size analyses would be required; moreover we required that other break location analyses be performed in order to fulfill the requirements of 10 CFR Part 50, §50.46 to determine the limiting break location and size. These additional analyses were provided in conformance to the minimum break spectrum and location requirements of 10 CFR Part 50, §50.46 and they confirmed that the recirculation line break is the limiting break location and, as stated earlier, 0.35 ft^2 is the limiting break area.

For the case of a postulated break in the core spray line, operating procedures are required to accommodate single failures in the onsite power supply for the other intact core spray system. In the event of failure of the diesel generator to supply power to the intact core spray loop pumps, there are approximately nine minutes in which core spray must be established. For the Oyster Creek system, the licensee has proposed that core spray can be established in less than nine minutes by manual action to disconnect the diesel generator associated with the broken loop from that loop and to connect it to the intact core spray loop in lieu of the failed diesel normally connected to the intact loop.

JCP&L has indicated that this will satisfy the need to establish spray in less than nine minutes since:

a. The system is designed to alert the operator to this specific event. Specific visual and audible alarms based on pressure sensors in the core spray lines and in the reactor vessel signal a "Core spray system 1 pipe break" or a "core spray system 2 pipe break".

b. Concise operational procedures are available and operator actions are quickly accomplished. Indication of "core spray system pipe break" calls for immediate verification that the other system is operational. If the second spray system's diesel is not operational, then procedures call for immediate connection of the broken spray system's diesel to the intact core spray system's bus.

On April 28, 1975, this procedure was "walked through" at Oyster Creek Station and the operator was able to accomplish the actions necessary to get core spray flow to the core in less than five minutes. While considering the urgency of such an emergency situation, this practical demonstration provided assurance that the needed actions to provide design core spray flow in nine minutes or less is possible because (1) the operator has clear indication of a problem, (2) concise procedures are available to compensate for a failed diesel generator, and (3) the required actions and procedures can be accomplished from the control room.

These procedures provided sufficient assurance that single failures of the diesel generators powering the core spray system will not adversely affect cooling system performance.

In our review, we also considered the effects of the worst electrical single failure, which would be a direct short to ground of the main electric bus which routes power to the core spray pumps of the unbroken loop. Such a failure would prevent the core spray pumps of the unbroken loop from operating. In this case, the core spray pumps of the unbroken loops cannot be rapidly connected to the power available from the operative bus and diesel generator which are aligned with the pumps of the broken loop.

A short of this type occurred during the early stages of the present refueling outage in March 1975. It has since been repaired. However, we have discussed this matter with JCP&L and have informed them of the need to revise the design of the electrical system to accommodate this short circuit passive failure in the electrical system which can adversely affect the ECC system.

JCP&L has indicated that it will submit for review and approval within 30 days a proposed modification to the electrical system to accommodate such a failure.

After approval, JCP&L has agreed to install such approved modifications within 30 days. It will also systematically review all elements of the electrical system associated with ECCS performance to update and verify that no other single passive electrical failure will adversely affect the ECC system.

In the interim event for this postulated set of circumstances, (core line break, loss of offsite power and short circuit of the diesel generator emergency bus supplying power to the unbroken core spray loop), sufficient cooling flow can be supplied through the condensate pumps to maintain clad temperatures (clad oxidation and metal water reaction) within the ECCS evaluation results. The condensate pumps which can be powered from the intact emergency diesel bus provide sufficient flow from the hotwell of the main condensers through the feedwater lines into the core to maintain adequate core cooling. The pumps can be energized and the valve connections to provide required flow can be made from the control room. After the core has been recovered, through use of the condensate pumps, there are a number of systems available for providing adequate long term cooling.

The condensate pumps that would be used for this backup purpose have been assessed at the time of the operating license review for seismic integrity, and can withstand these effects without loss of safety function.

Although such manual reconnections are not as desirable as automatic connection for a permanent resolution, they are sufficient for the interim period until the electrical system modifications can be accomplished.

Summary

We have reviewed the evaluation of ECCS performance submitted by JCP&L for the Oyster Creek Nuclear Generating Station and conclude that the evaluation has been performed wholly in conformance with the requirements of 10 CFR Part 50, §50.46 (a). Therefore, operation of the reactor would meet the requirement and criteria of 10 CFR §50.46 provided that operation is limited to the maximum average planar linear heat generation rates (MAPLHGR) of figure 3.10.1 and Section 3.10.A. of the April 30, 1975 Technical Specification Change Request No. 36 (Revision 1), and provided the reactor is operated in conformance to the assembly averaged power-void relationship Technical Specification given in section 3.10.C of the proposed Technical Specifications (Change Request No. 36, Revision 1).

An evaluation was not provided for ECCS performance during reactor operation with recirculation loops out of service. Therefore reactor operation under such conditions will not be authorized until the necessary analyses have been performed, evaluated and determined acceptable.

The facility can accommodate single failures in conformance with the requirements of 10 CFR §50.46 although certain manual actions discussed above are necessary to accommodate certain failures until system modifications discussed above are completed.

Conclusion

We have concluded, based on the considerations discussed above that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) 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.

Date: MAY 24 1975

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-219

JERSEY CENTRAL POWER & LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO PROVISIONAL OPERATING LICENSE

Notice is hereby given that the U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 8 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, Unit 1, located in Ocean County, New Jersey.

The amendment incorporates operating limits in the Technical Specifications for the facility based on an acceptable evaluation model that conforms with the requirements of Section 50.46 of 10 CFR Part 50.

The applications for the amendment comply 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. Notice of Proposed Issuance of Amendment to Provisional Operating License in connection with this action was published in the FEDERAL REGISTER on April 4, 1975 (40 F.R. 15137). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

For further details with respect to this action, see (1) the applications for amendment dated March 25 and 29, 1975, (2) Amendment No. 8 to License No. DPR-16, with Change No. 24, (3) the Commission's related Safety Evaluation, and (4) the Environmental Impact Appraisal. 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, 15 Hooper Avenue, Toms River, New Jersey 08753.

A copy of items (2), (3) and (4) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Reactor Licensing.

Dated at Bethesda, Maryland, this 24th day of May, 1975

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

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