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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. 16 to Provisional Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station. The amendment consists of changes to the Technical Specifications in response to your applications dated October 20, 1975 (supplemented by February 4, 1976 letter and revised by February 9, 1976 letter) and June 18, 1976 (supplemented by June 30, 1976 letter).

This amendment involves changes to the Technical Specifications minimum critical power ratio (MCPR) safety and operating limits, an increase in Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) and changes in selected surveillance requirements and remedial actions related to these limits.

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

Sincerely,

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

Enclosures:
1. Amendment No. 16
2. Safety Evaluation
3. Federal Register Notice

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DATE ➤	7/14/76	7/15/76	7/22/76	7/23/76	7/22/76	7/15/76

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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, UNIT NO. 1

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 16
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 October 20, 1975 (supplemented by February 4, 1976 letter and revised by February 9, 1976 letter) and June 18, 1976 (supplemented by June 30, 1976 letter), 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.

   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 a change to the Technical Specifications as indicated in the attachment to this license amendment.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Karl R. Goller

Karl R. Goller, Assistant Director
for Operating Reactors
Division of Operating Reactors

Attachment:
Changes to the
Technical Specifications

Date of Issuance: July 26, 1976

ATTACHMENT TO LICENSE AMENDMENT NO. 16

TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-16

DOCKET NO. 50-219

Replace pages 2.1-2, 2.1-4, 2.1-5, Figure 2.1-1, 2.3-1, 2.3-2, 2.3-3, 2.3-4, 2.3-8, 3.1-1, 3.1-12, Figure 3.10-1 and 4.10-1 with the attached revised pages. Add pages 3.10-6, 3.10-7, and 4.10-2. Pages 3.10-1 through 3.10-5 are issued in their entirety.

D. Whenever the reactor is in the shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than 4'-8" above the top of the normal active fuel zone.

E. The existence of a minimum critical power ratio (MCPR) less than 1.32 for 7 x 7 fuel and 1.34 for 8 x 8 fuel shall constitute violation of the fuel cladding integrity safety limit.

Bases:

The fuel cladding represents one of the primary physical barriers which separate radioactive material from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use-related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative, continuously measurable and tolerable. Fuel cladding perforations, however, could result from thermal effects if reactor operation is significantly above design conditions and the associated protection system set point. While fission product migration from cladding perforation is just as measurable as that from use-related cracking, the thermally-caused cladding perforations signal a threshold, beyond which still greater thermal conditions may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding safety limit is defined in terms of the reactor operating conditions which may result in cladding perforation.

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 margin to boiling transition is calculated from plant operating parameters such as core pressure, core flow, feedwater temperature, core power, and core power distribution. The margin for each fuel assembly is characterized by the critical power ratio (CPR) which is the ratio of the bundle power which would produce onset of transition boiling divided by the actual bundle power. The minimum value of this ratio for any bundle in the core is the minimum critical power ratio (MCPR)(10).

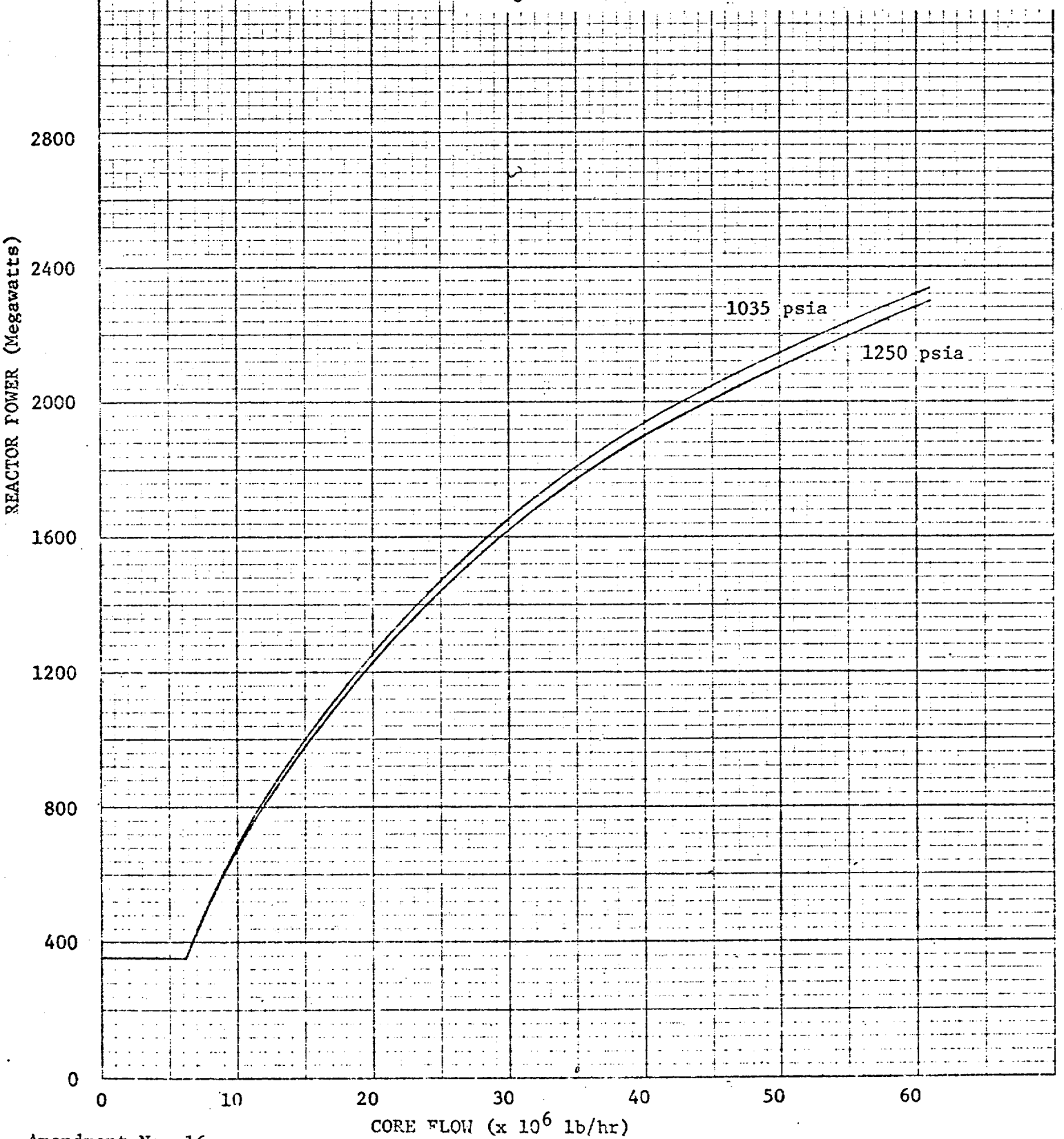
The safety limit curves shown in Figure 2.1.1 represent conditions which assure with better than 95 percent confidence a 95 percent probability of avoiding a critical heat flux occurrence. The critical power value was determined using the design basis critical power correlation given in reference 1. The operating range with MCPR >1.32 for 7 x 7 fuel and 1.34 for 8 x 8 fuel is below and to the right of these curves.

FIGURE 2.1.1 (June 76)

FUEL CLADDING INTEGRITY SAFETY LIMIT

NOTES:
1. Rated Power = 1930 MWt
2. Rated Flow = 61.0 x 10^6 lb/hr
3. Peaking Factors ≤Specification Values (PF_o)*
4. Core Pressure ≥600 psia
5. Reactor Water Level ≥10 ft.7in. above the top of the active fuel.

*For peaking factors greater than Specification Values (PF_o) see Specification 2.1.A.2.



The range in pressure used for Specification 2.1.A in the calculation of the fuel cladding integrity safety limit is from 600 to 1250 psia. Specification 2.1.B provides a requirement on power level when operating below 600 psia or 10% flow. In general, Specification 2.1.B will only be applicable during startup or shutdown of the plant. A review of all the applicable low pressure and low flow data (6,7) has shown the lowest data point for transition boiling to have a heat flux of 144,000 BTU/hr-ft^2. To insure applicability to the BWR fuel rod geometry, and provide a margin, a factor of one half was used, giving a critical heat flux of 72,000 BTU/hr-ft^2. This is equivalent to a core average power of 354 Mwt (18.3% of rated). This value is applicable to ambient pressure and no flow conditions. For any greater pressure or flow conditions there is increased margin.

During transient operation, the heat flux (thermal power-to-water) would lag behind the neutron flux due to the inherent heat transfer time constant of the fuel of 8-9 seconds. Also, the limiting safety system scram settings are at values which will not allow the reactor to be operated above the safety limit during normal operation or during other plant operating situations which have been analyzed in detail (2,3,4,8,9,10).

If the scram occurs such that the neutron flux dwell time above the limiting safety system setting is less than 1.75 seconds, the safety limit will not be exceeded for normal turbine or generator trips, which are the most severe normal operating transients expected. Following a turbine or generator trip, if it is determined that the bypass system malfunctioned, analysis of plant data will be used to ascertain if the safety limit has been exceeded, according to Specification 2.1.A. The dwell time of 1.75 seconds in Specification 2.1.C provides increased margin for less severe power transients.

Should a power transient occur, the event recorder would show the time interval the neutron flux is over its scram setting. When the event recorder is out of service, a safety limit violation will be assumed if the neutron flux exceeds the scram setting and control rod scram does not occur. The event recorder shall be returned to an operable condition as soon as practical.

During periods when the reactor is shutdown, considerations must also be given to water level requirements due to the effect of decay heat. Specification 2.1.D provides a limit for the shutdown water level. If reactor water level should drop below the top of the active fuel during shutdown conditions, the ability to cool the core is reduced. This reduction in core cooling capability could lead to elevated cladding temperatures and clad perforation. With a water level above the top of the active fuel, adequate cooling is maintained and the decay heat can easily be accommodated.

The lowest point at which the water level can be monitored is 4'-8" above the top of the active fuel. This is the low-low-low water level trip point. The safety limit has been established at 4'-8" to provide a point which can be monitored and also provide adequate margin.

REFERENCES

(1) XN-75-34, Revision 1, The XN-2 Critical Power Correlation, Exxon Nuclear Company, Inc., August 1, 1975.
(2) FDSAR, Volume I, Section 1.5-6.
(3) Licensing Application Amendment 28, Question III.A-12.
(4) Licensing Application Amendment 32, Question 13.
(5) FDSAR, Volume I, Section SII-7.3.
(6) E. Janssen, "Multirod Burnout at Low Pressure", ASME Paper 62-HT-26, August 1962.
(7) K. M. Becker, "Burnout Conditions for Flow of Boiling Water in Vertical Rod Clusters", AE-74 (Stockholm, Sweden), May 1962.
(8) License Application Amendment No. 55, Section 4.
(9) Licensing Application Amendment 65, Sections B.IV, B.VIII, and B.XI.
(10) Licensing Application Amendment No. 76 (Supplement No. 4).
(11) Deleted

2.3 LIMITING SAFETY SYSTEM SETTINGS

Applicability: Applies to trip settings on automatic protective devices related to variables on which safety limits have been placed.

Objective: To provide automatic corrective action to prevent the safety limits from being exceeded.

Specification: Limiting safety system settings shall be as follows:

FUNCTION	LIMITING SAFETY SYSTEM SETTINGS
1) Neutron Flux, Scram	
a) APRM	For recirculation flow, $W \leq 61 \times 10^6$ lb/hr:
	$\leq ([1.34 \times 10^{-6}]\ W + 34.0)$ percent of rated neutron flux when total peaking factors in all fuel types are less than or equal to those in Specification 2.1.A.1, or
	The lowest value of:
	$\leq ([1.34 \times 10^{-6}]\ W + 34.0) \left[\frac{PF_o}{PF}\right]$ percent of rated neutron flux from among those calculations for each fuel type with total peaking factors, $PF > PF_o$, where PF_o = peaking factor in Specification 2.1.A.1.
	For recirculation flow, $W > 61 \times 10^6$ lb/hr:
	≤ 115.7 percent of rated neutron flux when total peaking factors in all fuel types are less than or equal to those in Specification 2.1.A.1, or
	The lowest value of $\leq 115.7 \left[\frac{PF_o}{PF}\right]$ percent of rated neutron flux from among those calculations for each fuel type with total peaking factors $PF > PF_o$, where PF_o = peaking factor in Specification 2.1.A.1.
b) IRM	≤ 15 percent of rated neutron flux

FUNCTION	LIMITING SAFETY SYSTEM SETTINGS
2) Neutron Flux, Control Rod Block a)	For recirculation flow, $W \leq 61 \times 10^6$ lb/hr: $\leq ([1.34 \times 10^{-6}] W + 24.3)$ percent of rated neutron flux when total peaking factors in all fuel types are less than or equal to those in Specification 2.1.A.1, or The lowest value of: $\leq ([1.34 \times 10^{-6}] W + 24.3) \left[\frac{PF_o}{PF}\right]$ percent of rated neutron flux from among those calculated for each fuel type with total peaking factors, $PF > PF_o$, where PF_o = peaking factor in Specification 2.1.A.1 For recirculation flow, $W > 61 \times 10^6$ lb/hr: ≤ 106 percent of rated neutron flux when total peaking factors in all fuel types are less than or equal to those in Specification 2.1.A.1, or The lowest value of $\leq 106 \left[\frac{PF_o}{PF}\right]$ percent of rated neutron flux from among those calculated for each fuel type with total peaking factors, $PF > PF_o$, where PF_o = peaking factor in Specification 2.1.A.1.
3) Reactor High Pressure, Scram	$\leq$1060 psig.
4) Reactor High Pressure, Relief Valves Initiation	$\leq$1070 psig.
5) Reactor High Pressure, Isolation Condenser Initiation	$\leq$1060 psig with time delay $\leq$15 seconds.
6) Reactor High Pressure, Safety Valve Initiation	4 @ 1212 psig 4 @ 1221 psig $\pm$ 12 psi 4 @ 1230 psig 4 @ 1239 psig

FUNCTION	LIMITING SAFETY SYSTEM SETTINGS
7) Low Pressure Main Steam Line, MSIV CLosure	≥825 psig
8) Main Steam Line Isolation Valve Closure, Scram	≤10% Valve Closure from full open
9) Reactor Low Water Level, Scram	≥11'5" above the top of the active fuel as indicated under normal operating conditions.
10) Reactor Low-Low Water Level, Main Steam Line Isolation Valve Closure	≥7'2" above the top of the active fuel as indicated under normal operating conditions.
11) Reactor Low-Low Water Level, Core Spray Initiation	≥7'2" above the top of the active fuel.
12) Turbine Trip Scram	10 percent turbine stop valve(s) closure from full open.
13) Generator Load Rejection Scram	Initiate upon loss of oil pressure from turbine acceleration relay.

Bases: Safety limits have been established in Specifications 2.1 and 2.2 to protect the integrity of the fuel cladding and reactor coolant system barriers. Automatic protective devices have been provided in the plant design to take corrective action to prevent the safety limits from being exceeded in normal operation or operational transients caused by reasonable expected single operator error or equipment malfunction. This Specification establishes the trip settings for these automatic protection devices.

The Average Power Range Monitor, APRM(1), trip setting has been established to assure never reaching the fuel cladding integrity safety limit. The APRM system responds to changes in neutron flux. However, near rated thermal power the APRM is calibrated, using a plant heat balance, so that the neutron flux that is sensed is read out as percent of rated thermal power. For slow maneuvers, those where core thermal power, surface heat flux, and the power transferred to the water follow the neutron flux, the APRM will read reactor thermal power. For fast transients, the neutron flux will lead the power transferred from the cladding to the water due to the effect of the fuel time constant. Therefore, when the neutron flux increases to the scram setting, the percent increase in heat flux and power transferred to the water will be less than the percent increase in neutron flux.

The APRM trip setting will be varied automatically with recirculation flow with the trip setting at rated flow 61.0 x 10^5 lb/hr or greater being 115.7% of rated neutron flux. Based on a complete

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 fixed point scram setting so that all thermal limits are satisfied (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 exceeded 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 types I, II, and III; 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.

REFERENCES (Cont'd.)

(6) Licensing Application Amendment 11, Question V-9.
(7) License Application Amendment 76, Supplement No. 1.
(8) License Application Amendment 65, Section B.XI.
(9) License Application Amendment 69, Section III-D-5.
(10) License Application Amendment 65, Section B.IV.
(11) License Application Amendment 65, Section B.IX.
(12) License Application Amendment 76, Supplement No. 3, Section 2.0.
(13) License Application Amendment 76, Supplement No. 4.

LIMITING CONDITIONS FOR OPERATION

3.1 PROTECTIVE INSTRUMENTATION

Applicability: Applies to the operating status of plant instrumentation which performs a protective function.

Objective: To assure the operability of protective instrumentation.

Specifications: A. The following operating requirements for plant protective instrumentation are given in Table 3.1.1:

1. The reactor mode in which a specified function must be operable including allowable bypass conditions.

2. The minimum number of operable instrument channels per operable trip system.

3. The trip settings which initiate automatic protective action.

4. The action required when the limiting conditions for operation are not satisfied.

B. 1. Failure of four chambers assigned to any one APRM shall make the APRM inoperable.

2. Failure of two chambers from one radial core location in any one APRM shall make that APRM inoperable.

C. 1. 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 may not contain a combination of more than three (3) inoperable detectors (i.e., APRM channel failed or bypassed, or LPRM detectors failed or bypassed) out of the four (4) detectors located in either the A and B, or the C and D levels.

2. A Travelling In-Core Probe (TIP) chamber may be used as an APRM input to meet the criteria of 3.1.B and 3.1.C.1, provided the TIP is positioned in close proximity to one of the failed LPRM's. If the criteria of 3.1.B.2 or 3.1.C.1 cannot be met, power operation may continue at up to rated power level provided a control rod withdrawal block is operating or at power levels less than 61% of rated power until the TIP can be connected, positioned and satisfactorily tested, as long as Specification 3.1.B.1 and Table 3.1.1 are satisfied.

TABLE 3.1.1 (CONT'D.)

* Action required when minimum conditions for operation are not satisfied. Also permissible to trip inoperable trip system. When necessary to conduct tests and calibrations, one channel may be made inoperable for up to one hour per month without tripping its trip system.

** See Specification 2.3 for Limiting Safety System Settings.

Notes:

a. Permissible to bypass, with control rod block, for reactor protection system reset in refuel mode.

b. Permissible to bypass below 600 psig in refuel and startup modes.

c. One (1) APRM in each operable trip system may be bypassed or inoperable provided the requirements of specification 3.1.C and 3.10.D are satisfied. Two APRM's in the same quadrant shall not be concurrently bypassed except as noted below:

Any one APRM may be removed from service for up to one hour for test or calibration without inserting trips in its trip system only if the remaining operable APRM's meet the requirements of specification 3.1.B.1 and no control rods are moved outward during the calibration or test. During this short period, the requirements of specifications 3.1.B.2, 3.1.C and 3.10.D need not be met.

When in the Refuel Mode, two APRM's in the same quadrant may be made inoperable during replacement of an LPRM assembly, provided that the Source Range Channel and both Intermediate Range Channels in that quadrant are operable and provided that the Removable Jumpers for Refueling Non-Coincidence have been removed.

d. The (IRM) shall be inserted and operable until the APRM's are operable and reading at least 2/150 full scale.

e. Air ejector isolation valve closure time delay shall not exceed 15 minutes.

f. Unless SRM chambers are fully inserted.

g. Not applicable when IRM on lowest range.

h. One instrument channel in each trip system may be inoperable provided the circuit which it operates in the trip system is placed in a simulated tripped condition. If repairs cannot be completed within 72 hours the reactor shall be placed in the cold shutdown condition. If more than one instrument channel in any trip system becomes inoperable the reactor shall be placed in the cold shutdown condition. Relief valve controllers shall not be bypassed for more than 8 hours (total time for all controllers) in any 30-day period and only one relief valve controller may be bypassed at a time.

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 7x7 fuel) and 14.5 KW/ft (for 8x8 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:

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

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	$LHGR_d$	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

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 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.

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
PR = 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
I, II, III	.365
IIIE, IIIF	.377
V, VB	.332

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

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

Non-jet pump BWR ECCS models utilize an emperical correlation to determine the duration 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 evaluted 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.

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.

For transient operation up to the fuel cladding integrity safety limit protection is provided against a MCPR of 1.34 for 8x8 fuel and 1.32 for 7x7 fuel. The actual steady-state operating power level provides margin to this limit by an amount corresponding to the maximum decrease in CPR resulting from single operator error or equipment malfunction from a steady-state level.

These resulting operating MCPR limits combined with the transient analysis results provide assurance that the fuel cladding integrity safety limit will not be violated during anticipated operating transients.

The ARPM response is used to predict when the rod block occurs in the analysis of the rod withdrawal error transient. The transient rod position at the rod block and corresponding MCPR can be determined. The MCPR has been evaluated for different APRM responses which would result from changes in the APRM

status as a consequence of an APRM channel bypasses and/or LPRM input failed or bypassed. The results indicate that the steady state MCPR required to protect the minimum transient MCPR of 1.34 at the rod block ranges from 1.5 to 1.6 depending on the APRM system status. (4)

In order to provide for a limit which is considered to be bounding to future operating cycles, the variable limits have been conservatively adjusted upward to range from 1.52 to 1.64.

The time interval of eight (8) hours to adjust the steady state MCPR to account for a degradation in the APRM status is justified on the basis of instituting a control rod block which precludes the possibility of experiencing a rod withdrawal error transient since rod withdrawal is physically prevented. This time interval is adequate to allow the operator to either increase the MCPR to the appropriate value or to upgrade the status of the APRM system while in a condition which prevents the possibility of this transient occurring.

REFERENCES

(1) Oyster Creek Nuclear Generating Station, Loss-of-Coolant Accident Analysis Reevaluation and Technical Specification Change Request No. 42, Attachment I, dated December 23, 1975.

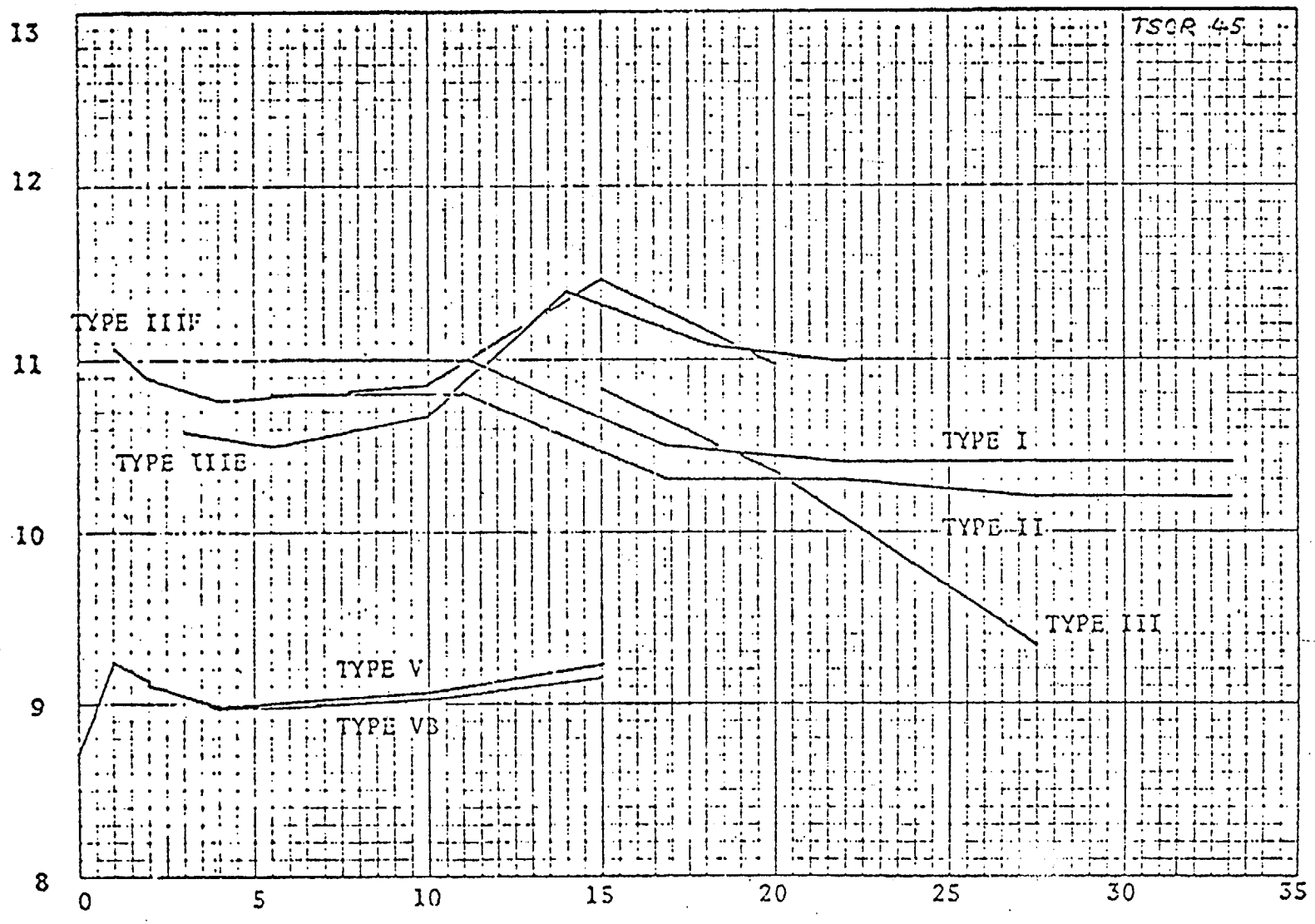
(2) XN-75-36NP, "Spray Cooling Heat Transfer Phase I Test Results ENC-8x8 BWR Fuel 60 and 63 Active Rods".

(3) XN-75-36NP, Supplement 1, "Spray Cooling Heat Transfer Phase I Test Results ENC - 8x8 BWR Fuel 60 and 63 Active Rods".

(4) Oyster Creek Nuclear Generating Station Amendment No. 76 (Supplement No. 4), Section 21, dated October 20, 1975.

FIGURE 3.10-1

MAXIMUM ALLOWABLE AVERAGE PLANAR
LINEAR HEAT GENERATION RATE



Amendment No. 16

AVERAGE PLANAR EXPOSURE
(GWD/MTM)

4.10 ECCS RELATED CORE LIMITS

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

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

Specification: A. Average Planar LHGR

The APLHGR for each type of fuel as a function of average planar exposure shall be checked daily during reactor operation at $\geq$ 25% rated thermal power.

B. Local LHGR

The LHGR as a function of core height shall be checked daily during reactor operator $\geq$ 25% rated thermal power.

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 least every 72 hours thereafter during power operation.

D. Minimum Critical Power Ratio (MCPR)

MCPR and APRM status shall be checked daily during reactor power operation at $\geq$ 25% rated thermal power.

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 veritife 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 relstionship is not violated. Additionally reduced power operation involves less stored 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.

The minimum critical power ratio (MCPR) is unlikely to change significantly during steady state power operation so that 24 hours is an acceptable frequency for surveillance. The 24 hour frequency is also acceptable for the APRM status check since neutron monitoring system failures are infrequent and a downscale failure of either an APRM or LPRM initiates a control rod withdrawal block thus precluding the possibility of a control rod withdrawal error.

At core power levels less than or equal to 25% rated thermal power the reactor will be operating at or above the minimum recirculation pump speed. For all designated control rod patterns which may be employed at this point, operating plant experience and thermal hydraulic analysis indicate that the resulting APLHGR, LHGR, and MCPR values all have considerable margin to the limits of section 3.10. Consequently, monitoring of these quantities below 25% of rated thermal power is not required.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

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

JERSEY CENTRAL POWER AND LIGHT COMPANY

DOCKET NO. 50-219

OYSTER CREEK NUCLEAR GENERATING STATION

Introduction

By letter dated October 20, 1975, a supplement dated February 4, 1976 and a revision (Revision No. 1) dated February 9, 1976, the Jersey Central Power and Light Company requested an amendment to Appendix A of the Oyster Creek Nuclear Generating Station Provisional Operating License No. DPR-16. Additional information was provided by JCP&L letters dated April 15, 1976 (response to questions raised by NRC in a letter dated March 17, 1976), and May 6, 1976 (information on the present operating cycle).

By letter dated June 18, 1976 and a supplement dated June 30, 1976, JCP&L requested an additional amendment involving changes to the Technical Specifications to (1) state explicit remedial action to be taken in the event that Maximum Average Planar Linear Heat Generation Rate (MAPLHGR), Maximum Local Linear Heat Generation Rates (MLLHGR), or the MCPR is exceeded, (2) eliminate requirements to monitor MAPLHGR, MLLHGR and MCPR at power levels less that 25% of rated power level (3) increase the frequency of checking MCPR from once every 72 hours to daily and (4) increase MAPLHGR for types V and VB fuel.

Both proposed amendments have been considered in the staff safety evaluation and have been combined for one amendment of the operation license.

Background

A topical report "The XN-2 Critical Power Correlation", XN-75-34, dated August 1, 1975 was submitted to the NRC by Exxon Nuclear Company Inc. The report is based upon the application of the XN-2 CHF correlation for predicting the power to boiling transition and is to be used by EXXON in the design and operation of (reload) cores for boiling water reactors (BWR's) over the range of applicability of the correlation. NRC has reviewed and approved the use of the XN-2 critical power correlation.

Using the new XN-2 critical power correlation, JCP&L has recalculated new MCPR safety and operating limits. The proposed operating limits allow for operating transients and, for the first time, account for various levels of degradation of the average power range monitors (APRM) and also future operating cycles to assure that the MCPR safety limit is not exceeded.

As proposed by JCP&L in their response to NRC letter dated February 9, 1976, the explicity remedial action to be taken by the operator in the event that technical specification operating limits for MAPLHGR, MLLHGR or MCPR are exceeded, is essentially the same as the remedial action at other nuclear plants that we have reviewed and accepted. The proposed remedial action is therefore acceptable for Oyster Creek. Moreover, the requirements for monitoring MAPLHGR, MLLHGR and MCPR as well as the frequency for checking MCPR are also the same as at other nuclear power plants that we have reviewed and accepted. Hence, these proposed technical specification changes, on this basis, are also acceptable for the Oyster Creek Nuclear Generating Station.

The 8x8 fuel assembly spray cooling tests, conducted by Exxon Nuclear Company (ENC) have provided data to confirm that the "Appendix K" heat transfer coefficients for 7x7 fuel assemblies are valid for the ENC 8x8 fuel assemblies. Until these tests were completed, an allowance for the uncertainty was required and accordingly, the ECCS performance evaluation was based on heat transfer coefficients that were only 90% of the values determined for the 7x7 fuel assembly. The test results are documented in the ENC report "Spray Cooling Heat Transfer Phase I Test Results - ENC 8x8 BWR Fuel 60 and 63 Active Rods", XN-75-36, and Supplement 1 dated August 8, 1975.

Discussion and Evaluation

The staff has verified that Oyster Creek core thermal hydraulic conditions fall within the range of operating conditions used in the XN-2 analyses, and concludes that the XN-2 critical power correlation can be used for the analysis of boiling transition in the Oyster Creek core.

The results of analyses using the XN-2 critical power correlation indicate that a minimum critical power ratio (MCPR) of 1.32 assures with 95 percent confidence that there is more than 95 percent probability of avoiding boiling transition. This safety limit MCPR applies to both 7x7 and 8x8 fuel. However, it was determined that a 1% increase in MCPR is required to account for the four inert rods in 8x8 fuel assemblies. Thus, the safety limit MCPR is taken as 1.34 for the 8x8 fuel. It is concluded that the proposed fuel cladding integrity - MCPR safety limits of 1.32 for 7x7 fuel and 1.34 for 8x8 fuel (with four inert rods) are acceptable to prevent degraded heat transfer conditions. MCPR operating limits are established in order that, during abnormal operational transients, the MCPR's remain above the safety limits and, therefore, nucleate to film boiling transition is avoided if such transients should occur. The MCPR operating limits in effect provides additional margin to avoid boiling transition during normal steady state conditions thereby providing greater assurance of fuel clad integrity.

To assure that the safety limit MCPR is not exceeded during anticipated abnormal operation transients, the most limiting transients have been re-analyzed using XN-2 critical power correlation to determine the change in MCPR (Δ MCPR) during the transients. The transients re-analyzed for the previous Cycle 5 core configuration were turbine trip without bypass, trip of five recirculation pumps, slow loss of feedwater heater, and control rod withdrawal error transient. Of these four transients, control rod withdrawal error is the most limiting. If all APRM's are operable and no LPRM inputs have been bypassed, a rod block occurs when the transient rod is out 2-1/2 feet, and the ΔMCPR is 0.16 for Cycle 5 and 0.19 for Cycle 6. In these cases the calculated operational MCPR's are .50 and .51 at rated power and less than the limit of 1.52 shown below.

If some of the APRM's are inoperable, or if some LPRM's are bypassed, the rod block occurs later and the MCPR is greater. The new MCPR limits with allowance for the different conditions of APRM and LPRM availability are listed below:

	MCPR Safety Limit:
7x7 fuel assemblies	1.32
8x8 fuel assemblies	1.34

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

With respect to the increased MAPLHGR for types V and VB fuel shown in revised figure 3.10-1 of the technical specifications, the staff has concluded that the Exxon Nuclear Company Topical Reports "Spray Cooling Heat Transfer Phase 1 Test Results ENC - 8x8 BWR 60 and 63 Active Rods" and Supplements dated September 1975, and October 1975 that were approved by NRC on May 17, 1976,(1) are an acceptable reference to justify use of the full "Appendix K" specified spray coefficients for ENC 8x8 fuel in the Oyster Creek reactor.

We agree with the Jersey Central Power and Light Company (JCP&L) statement that use of the new spray coefficients will not change the worst break location, size, or associated single failure(2). Therefore, reanalysis of the limiting 0.50 ft^2 break at the listed exposures(3) constitutes an acceptable ECCS analysis meeting all requirements of Appendix K to 10 CFR 50, and we approve operation of the Oyster Creek Reactor up to the Maximum Average Planan Linear Heat Generation Rate (MAPLHGR) limits resulting from those reanalyses utilizing the full Appendix K specified spray coefficients as shown in revised figure 3.10-1 of the technical specifications.

We have concluded that the proposed technical specification changes to include the new operating MCPRs and MAPLHGRs are acceptable as well as the other changes that have been identified above to place more specific requirements on the operator.

Environmental Consideration

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 statement, negative declaration, or environmental appraisal need not be prepared in connection with the issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the changes to not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the changes do not involve a significant hazards consideration, (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: July 26, 1976

(1) D. Eisenhut letter to Karl R. Goller reporting DOR review of above references 1 and 2, May 17, 1976.

(2) (letter due in 06/29/76 to V. Stello, attn. J. Shea)

(3) JCP&L letter to V. Stello, Technical Specification change request #46, June 18, 1976.

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. 16 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 involves changes to the Technical Specifications minimum critical power ratio (MCPR) safety and operating limits, an increase in Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) and changes in selected surveillance requirements and remedial actions related to these limits.

The applications for 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 Facility Operating License in connection with the change in MCPR was published in the FEDERAL REGISTER ON December 17, 1975 (40 FR 58518). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

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 statement, negative declaration or 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 applications for amendment dated October 20, 1975 (supplemented by February 4, 1976 letter and revised by February 9, 1976 letter) and June 18, 1976 (supplemented by June 30, 1976 letter), (2) Amendment No. 16 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, 15 Hooper Avenue, Toms River, New Jersey 08753.

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 26th day of July, 1976.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

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