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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 Amendment No. 9 to Provisional Operaing License No. DPR-16 for the Oyster Creek Nuclear Generating Station. The amendment includes Change No. 25 to the Technical Specifications in accordance with your applications dated May 31, 1974 and January 31, 1975.

This amendment authorized operation with 8x8 fuel assemblies and changes the Technical Specifications in order to: (1) reduce the maximum allowable in-sequence control rod reactivity worth, (2) require a greater degree of operability of the rod worth minimizer and (3) change related procedural controls during reactor startup.

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

Sincerely,

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

Enclosures:
1. Amendment No. 9
2. Safety Evaluation
3. Federal Register Notice

cc: See next page

CP (1)

OFFICE	ORB#3	ORB#3	OELD	AD:RL/ORs	RL	DR
SURNAME	WAPaulson/dg WAP	GLear GL	J. Scinto	KRGoller KRG	AGiambusso KRG for	VStello
DATE	5/24/75	5/24/75	5/24/75 GL	5/24/75	5/24/75	5/24/75

Form AEC-318 (Rev. 9-53) AECM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166

Jersey Central Power & Light Company

cc: w/enclosures

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UNITED STATES NUCLEAR REGULATORY COMMISSION

JERSEY CENTRAL POWER & LIGHT COMPANY

DOCKET NO. 50-219

OYSTER CREEK NUCLEAR GENERATING STATION

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 9
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 May 31, 1974, as revised January 30 and 31, 1975, and supplements dated March 25 and 29, April 24 and 30, and May 14 and 23, 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 application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations; 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 Paragraph 3.B. of Provisional Operating License No. DPR-16 is hereby amended to read as follows:

"B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised, 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. 25."

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. 25 to
the Technical Specifications

Date of Issuance: MAY 24 1975

ATTACHMENT TO LICENSE AMENDMENT NO. 9

CHANGE NO. 25 TO THE TECHNICAL SPECIFICATIONS

PROVISIONAL OPERATING LICENSE NO. DPR-16

DOCKET NO. 50-219

Replace pages 1.0-5, 2.1-1, 2.1-2, 2.1-3, 2.1-4, 2.1-5, 2.3-1, 2.3-2, 2.3-3, 2.3-4, 2.3-5, 2.3-8, 3.1-1,3.1-1a, 3.1-6, 3.1-6a, 3.1-12, 3.2-1 3.2-2, 3.2-4, 3.2-9, 3.10-1, 3.10-2, 3.10-4, 4.2-1a, 4.10-1 with the attached revised pages.

Replace Figure 2.1.1 with the attached revised figure.

Insert new pages 3.2-1a and 4.10-1a.

1.21 CORE ALTERATION

A core alteration is the addition, removal, relocation or other manual movement of fuel or controls in the reactor core. Control rod movement with the control rod drive hydraulic system is not defined as a core alteration.

1.22 MINIMUM CRITICAL POWER RATIO

25

The minimum critical power ratio is the ratio of that power in a fuel assembly which is calculated to cause some point in that assembly to experience boiling transition to the actual assembly operating power.

SECTION 2

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMIT - FUEL CLADDING INTEGRITY

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

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

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

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

Fuel Types I, II and III

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

Fuel Types IIIE and IIIF

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

For 8 x 8 Fuel

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

25

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

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

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

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

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

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 1.37 for 7 x 7 fuel and 1.40 for 8 x 8 fuel shall constitute violation of the fuel cladding integrity safety limit. 25

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). 25

The safety limit curves shown in Figure 2.1.1 represent the conditions for which there is greater than 99 percent confidence that the most limiting bundle as a minimum critical power ratio (MCPR) greater than 1.37 for 7 x 7 fuel and 1.40 for 8 x 8 fuel. The MCPR value was determined using the design basis critical heat flux correlation given in JN-72-18 (1). The operating range with MCPR > 1.37 for 7 x 7 fuel and 1.40 for 8 x 8 fuel is below and to right of these curves.

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

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

The power shape used in the calculation of Figure 2.1.1 is given in Table 3.2 of Reference 12 for a peak to average power of 1.5 with a peak location at the core midplane (X/L = 0.5). Table 3.2 further shows an axial power shape with an axial peak of the same magnitude but with a peak location above the core midplane (X/L = 0.65). These power shapes result in total peaking factors for each fuel type as shown in Specification 2.1.A.1. The total peaking factor for each fuel type is to be less than that specified in Section 2.1.A.1 at rated power. When operating below rated power with higher peaking factors as during control rod manipulation or near end of core life, applicability of the safety limit is assured by applying the reduction factors specified in 2.1.A.2. 25

The feedwater temperature assumed was the maximum design temperature output of the feedwater heaters at the given pressures and flows (e.g., 334°F at 1035 psia and 100% flow). For any lower feedwater temperature, subcooling is increased and the curves provide increased margin.

The water level assumed in the calculations was ten inches below the reactor low water level scram point (10'-7" above the top of the active fuel), which is the location of the bottom of the steam separator skirts. Of course, the reactor could not be operated in this condition. As long as the water level is above this point, the safety limit curves are applicable. As long as the water level is above the bottom of the steam separator skirts, the amount of carryunder would not be increased and the core inlet enthalpy would not be influenced.

The values of the parameters involved in Figure 2.1.1 can be determined from information available in the control room. Reactor pressure and flow are recorded and the APRM in-core nuclear instrumentation is calibrated in terms of percent power.

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).

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.

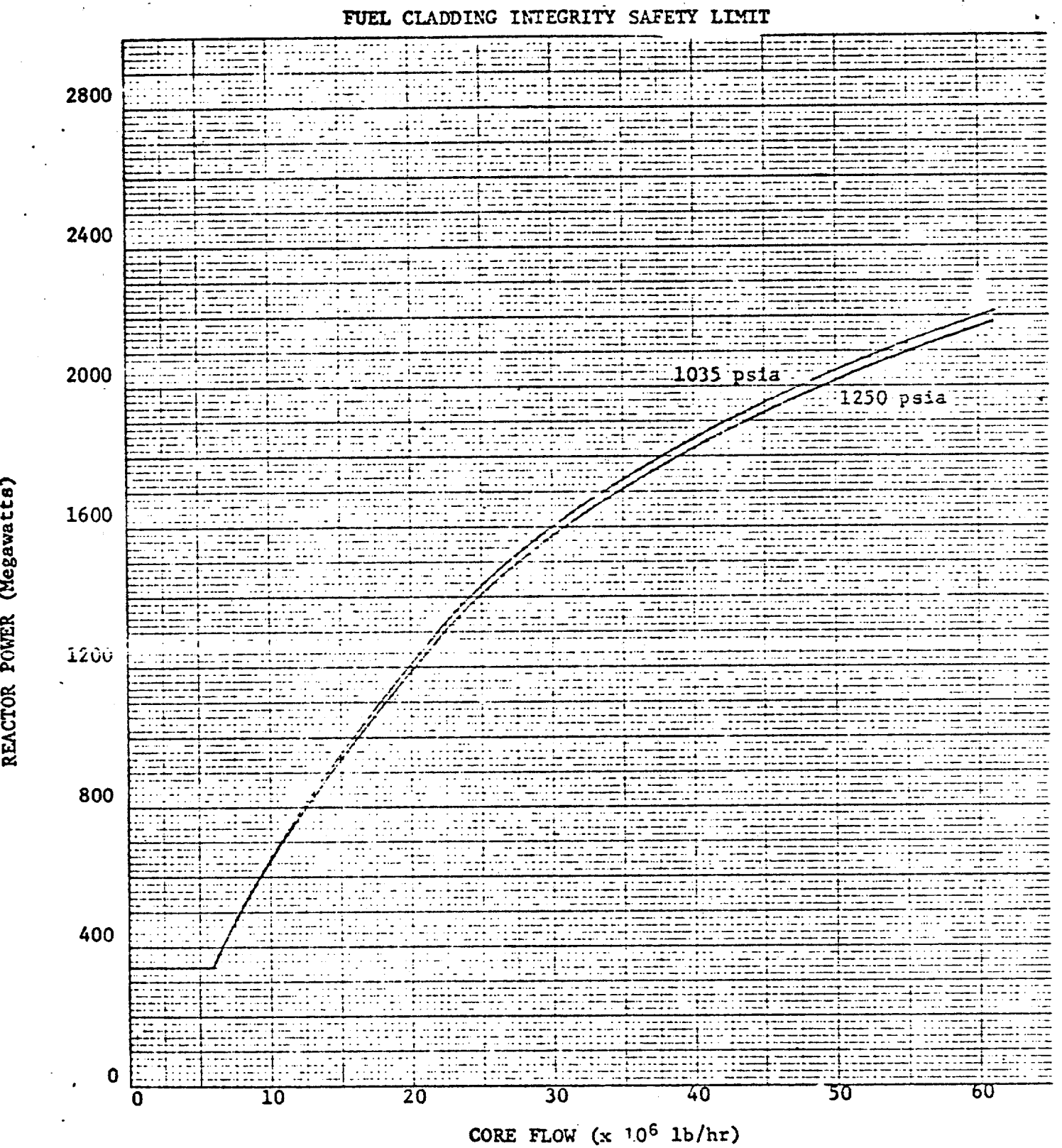
During periods when the reactor is shutdown, considerations must also be given to water level requirements due to the effect of decay heat. Specification 2.1.D provides a limit for the shutdown water level. If reactor water level should drop below the top of the active fuel during shutdown conditions, 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) L. H. Steves, A. M. Sutney, O. E. Fitzsimmons, "XN-1 Critical Heat Flux Correlation for Boiling Water Reactor Fuel"., JN-72-18, Revision 1, and License Application Amendment No. 76, Appendix A.
(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. 3). 25

FIGURE 2.1.1



NOTES:
1. Rated Power = 1930 MWt
2. Rated flow = 61.0 x 10^6 lb/hr
3. Peaking factors $\leq$ Specification Values (PF_o)*
4. Core pressure $\geq$ 600 psia
5. Reactor Water Level $\geq$ 10 ft. 7 in. above the top of of the active fuel.

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

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:	
	$\leq ([1.22 \times 10^{-6}]\ W + 35.58)$ 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	25
	The lowest value of: $\leq ([1.22 \times 10^{-6}]\ W + 35.58) \left[\frac{PF_o}{PF}\right]$ 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:	
	≤ 110 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	25
	The lowest value of $\leq 110 \left[\frac{PF_o}{PF}\right]$ 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) APRM	For recirculation flow, $W \leq 61 \times 10^6$ lb/hr: $\leq$ ([1.12×10^{-6}] W + 28.68) 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: $\leq$ ([1.12×10^{-6}] W + 28.68) $[\frac{PF_o}{PF}]$ 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 25 For recirculation flow, $W > 61 \times 10^6$ lb/hr: $\leq$ 97 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 $\leq 97 [\frac{PF_o}{PF}]$ 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. 25
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 10^6 lb/hr or greater being 110% of rated neutron flux. Based on a complete 25

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 (3,4,12). 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 the fixed 120% 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 ± 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 97% of rated power. For the single rod withdrawal error this setting causes rod block before MCPR reaches 1.37 for 7 x 7 fuel and 1.4 for 8 x 8 fuel(12). 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 = 1.37 for 7 x 7 fuel and 1.4 for 8 x 8 fuel even assuming there is no control rod block action(7). 25

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 Spec. 2.1.1) in the unusual event of higher peaking factors. Also, to insure MCPR's greater than 1.37 for 7 x 7 fuel and 1.40 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. 25

For operation in the startup mode while the reactor is at low pressure, the IRM scram setting of 15% of rated power provides 22% thermal margin between the maximum power and the safety limit, 18.3% of rated. The margin is adequate to accommodate anticipated maneuvers associated with power plant startup. There are a few possible sources of rapid reactivity input to the system in the low power low flow condition. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures backed up by the rod worth minimizer. Worth of individual rods is very low in a uniform rod pattern. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated, the rate of power rise is very slow. Generally the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than five percent of rated per minute, and the IRM system would be more than adequate to assure a scram before the power could exceed the safety limit. The IRM scram remains active until the mode switch is placed in the run position at which time the trip becomes a coincident IRM upscale, APRM downscale scram. The Reactor Protection System is designed such that reactor pressure must be above 825 psig to successfully transfer into the RUN mode, thus assuring protection for the fuel cladding safety limit.

The settings on the reactor high pressure scram, anticipatory scrams, reactor coolant system relief valves and isolation condenser have been established to assure never reaching the reactor coolant system pressure safety limit as well as assuring the system pressure does not exceed the range of the fuel cladding integrity safety limit. In addition, the APRM neutron flux scram and the turbine bypass system also provide protection for these safety limits, e.g., turbine trip and loss of electrical load transients (8). In addition to preventing power operation above 1060 psig, the pressure scram backs up the other scrams for these transients and other steam line isolation type transients. With the addition of the anticipatory scrams, the transient analysis for operation at 1930 MWt shows that the turbine trip with failure of the bypass system transient is the worst case transient with respect to peak pressure. Analysis of this transient shows that the relief valves limit the peak pressure to 1188 psig (10), well below the 1250 psig range of applicability of the fuel cladding integrity safety limit and the 1375 psig reactor coolant system pressure safety limit. Actuation of the isolation condenser during these transients removes the reactor decay heat without further loss of reactor coolant thus protecting the reactor water level safety limit.

REFERENCES (Cont'd.)

(6) Licensing Application Amendment 11, Question V-9.
(7) License Application Amendment 76, Supplement No. 1. 125
(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.

SECTION 3

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. 25

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 and 3.1.C.1 cannot be met, power operation may continue below 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. 25

Bases:

The plant protection system automatically initiates protective functions to prevent exceeding established limits. In addition, other protective instrumentation is provided to initiate action which mitigates the consequences of accidents or terminates operator control. This specification provides the limiting conditions for operation necessary to preserve the effectiveness of these instrument systems.

Table 3.1-1 defines, for each function, the minimum number of operable instrument channels for an operable trip system for the various functions specified. There are usually two trip systems required or available for each function. The specified limiting conditions for operation apply for the indicated modes of operation. When the specified limiting condition cannot be met, the specified actions required shall be undertaken promptly to modify plant operation to the condition indicated in a normal manner. Conditions under which the specified plant instrumentation may be out-of-service are also defined in Table 3.1.1.

period that it is required to monitor the neutron flux. This downscale rod block is also bypassed in IRM Ranges 8 and higher. It is not required at this power level since good indication exists in the Intermediate Ranges and the SRM will be reading approximately 5×10^5 CPS when using IRM Ranges 8 and higher.

The IRM downscale rod block in conjunction with the chamber full-in position and range switch setting, provides a rod block to assure that the IRM is in its most sensitive condition before startup. If the two latter conditions are satisfied, control rod withdrawal may commence even if the IRM is not reading at least 5%. However, after a substantial neutron flux is obtained, the rod block setting prevents the chamber from being withdrawn to an insensitive area of the core.

The APRM downscale setting of $\geq$2/150 full scale is provided in the run mode to prevent control rod withdrawal without adequate neutron monitoring.

High flow in the main steamline is set at 120% of rated flow. At this setting the isolation valves close and in the event of a steam line break limit the loss of inventory so that fuel clad perforation does not occur. The 120% flow would correspond to the thermal power so this would either indicate a line break or too high a power.

Temperature sensors are provided in the steam line tunnel to provide for closure of the main steamline isolation valves should a break or leak occur in this area of the plant. The trip is set at 50°F above ambient temperature at rated power. This setting will cause isolation to occur for main steamline breaks which result in a flow of a few pounds per minute or greater. Isolation occurs soon enough to meet the criterion of no clad perforation.

The low-low-low water level trip point is set at 4'8" above the top of the active fuel and will prevent spurious operation of the automatic relief system. The trip point established will initiate the automatic depressurization system in time to provide adequate core cooling.

Specification 3.1.B.1 defines the minimum number of APRM channel inputs required to permit accurate average core power monitoring. Specifications 3.1.B.2 and 3.1.C.1 further define the distribution of the operable chambers to provide monitoring of local power changes that might be caused by a single rod withdrawal. Any nearby, operable LPRM chamber can provide the required input for average core monitoring. A Travelling In-core Probe or Probes can be used temporarily to provide APRM input(s) until LPRM replacement is possible. Since APRM rod block protection is not required below 61% of rated power,[1] as discussed in Section 2.3, Limiting Safety System Settings, operation may continue below 61% as long as Specification 3.1.B.1

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and the requirements of Table 3.1.1 are met. In order to maintain reliability of core monitoring in that quadrant where an APRM is inoperable, it is permitted to remove the operable APRM from service for calibration and/or test provided that the same core protection is maintained by alternate means.

In the rare event that Travelling In-core Probes (TIPs) are used to meet the requirements 3.1.B or 3.1.C, the licensee may perform an analysis of substitute LPRM inputs to the APRM system using spare (non-APRM input) LPRM detectors and change the APRM system as permitted by 10 CFR 50.59. 25

Whenever it is necessary to replace an LPRM assembly, the operation requires the removal of fuel bundles in order to eliminate interference with the LPRM assembly. During the operation, the reactor mode switch will be locked in the REFUEL position in accordance with Technical Specification 3.9.B. In addition, the initial fuel loading non-coincidence jumpers in the Reactor Protective System will be removed. This provides additional protection for the core because any one out of four Source Range Monitor (SRM) channels, any one of eight APRM channels can produce a full scram (i.e., trip both Protection System Channels) if the flux reaches their respective setpoints.

Reference:

(1) NEDO-10189 "An Analysis of Functional Common Mode Failures in GE BWR Protection and Control Instrumentation", L. G. Frederick, et. al, July 1970.

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 Specification 3.1.C.1 is satisfied. Two APRM's in the same quadrant shall not be concurrently bypassed or inoperable with the following exceptions: 25

If one APRM in a quadrant cannot satisfy Technical Specification 3.1.B.2 or 3.1.C.1, the other APRM channel in that quadrant may be removed from service for up to one hour for test or calibration without inserting trips in its trip systems only if the first APRM is unbypassed and meets Technical Specification 3.1.B.1, and no control rod is moved outward during the calibration and/or test. 25

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.

3.2 REACTIVITY CONTROL

Applicability: Applies to core reactivity and the operating status of the reactivity control systems for the reactor.

Objective: To assure reactivity control capability of the reactor.

Specification: A. Core Reactivity

The core reactivity shall be limited such that the core could be made subcritical at any time during the operating cycle, with the strongest operable control rod fully withdrawn and all other operable rods fully inserted.

B. Control Rod System

1. The control rod drive housing support shall be in place during power operation and when the reactor coolant system is pressurized above atmospheric pressure with fuel in the reactor vessel, unless all control rods are fully inserted and Specification 3.2.A is met.

2. The Rod Worth Minimizer (RWM) shall be operable during each reactor startup until reactor power reaches 10% of rated power except as follows:

   (a) Should the RWM become inoperable after the first twelve rods have been withdrawn, the startup may continue provided that a second licensed operator verifies that the licensed operator at the reactor console is following the rod program.

   (b) Should the RWM be inoperable before a startup is commenced or before the first twelve rods are withdrawn, one startup during each calendar year may be performed without the RWM provided that the second licensed operator verifies that the licensed operator at the reactor console is following the rod program and provided that a Station Engineer from the Technical Group also verifies that the rod program is being followed. A startup without the RWM as described in this subsection shall be reported in a special report to the Nuclear Regulatory Commission within 30 days of the startup stating the reason for the failure of the RWM, the action taken to repair it and the schedule for completion of the repairs. 25

Control rod patterns shall be established so that the maximum worth of any in sequence control rod shall be less than 1.25% Δk, except:

1. During low power physics tests, and

2. During withdrawal of control rods which do not bring the reactor core to a critical condition. 25

3. The average of the scram insertion times of all operable control rods shall be no greater than:

Percent of Rod Length Inserted	Seconds
5	0.375
20	0.900
50	2.00
90	5.00

The average of the scram insertion times for the three fastest control rods of all groups of four control rods in a two by two array shall be no greater than:

Percent of Rod Length Inserted	Seconds
5	0.398
20	0.954
50	2.120
90	5.300

Any four rod group may contain a control rod which is

Any four rod group may contain a control rod which is valved out of service provided the above requirements and Specification 3.2.A are met. Time zero shall be taken as the de-energization of the pilot scram valve solenoids.

4. Control rods which cannot be moved with control rod drive pressure shall be considered inoperable. Inoperable control rods shall be valved out of service, in such positions that Specification 3.2.A is met. In no case shall the number of rods valved out of service be greater than six during power operation. If this specification is not met, the reactor shall be placed in the shutdown condition.

5. Control rods shall not be withdrawn for approach to criticality unless at least three source range channels have an observed count rate equal to or greater than 3 counts per second.

6. The control rod density shall be greater than 3.5 percent during all modes of reactor operation. 25

C. Standby Liquid Control System

1. The standby liquid control system shall be operable at all times when the reactor is not shutdown by the control rods such that Specification 3.2.A is met and except as provided in Specification 3.2.C.3.

2. The standby liquid control solution shall be maintained within the volume - concentration requirement area in Figure 3.2-1 and at a temperature not less than the temperature presented in Figure 3.2-2 at all times when the standby liquid control system is required to be operable.

3. If one standby liquid control system pumping circuit becomes inoperable during the run mode and Specification 3.2.A is met the reactor may remain in operation for a period not to exceed 7 days, provided the pump in the other circuit is demonstrated daily to be operable.

D. Reactivity Anomalies

The difference between an observed and predicted control rod inventory shall not exceed the equivalent of one percent in reactivity. If this limit is exceeded and the discrepancy cannot be explained, the reactor shall be brought to the cold, shutdown condition by normal orderly shutdown procedure. Operation shall not be permitted until the cause has been evaluated and appropriate corrective action has been completed. The AEC shall be notified within 24 hours of this situation in accordance with Specification 6.6.B.

Bases: Limiting conditions of operation on core reactivity and the reactivity control systems are required to assure that the excess reactivity of the reactor core is controlled at all times. The conditions specified herein assure the capability to provide reactor shutdown from steady state and transient conditions and assure the capability of limiting reactivity insertion rates under

which could be added by this small amount of rod withdrawal, which is less than a normal single withdrawal increment, will not contribute to any damage to the reactor coolant system. The support is not required when no fuel is in the core since no nuclear consequences could occur in the absence of fuel. The support is not required if the reactor coolant system is at atmospheric pressure since there would then be no driving force to rapidly eject a drive housing. The support is not required if all control rods are fully inserted since the reactor would remain subcritical even in the event of complete ejection of the strongest control rod(4).

The Rod Worth Minimizer(5) provides automatic supervision of conformance to the specified control rod patterns. It serves as a backup to procedural control of control rod worth. In the event that the RWM is out of service when required, a licensed operator can manually fulfill the control rod pattern conformance functions of the RWM in which case the normal procedural controls are backed up by independent procedural controls to assure conformance during control rod withdrawal. This allowance to perform a startup without the RWM is limited to once each calendar year to assure a high operability of the RWM which is preferred over procedural controls.

Control rod sequences are characterized by homogeneous, scattered patterns of withdrawn rods similar to that indicated in Figure 7-14 of Reference (12). The maximum rod strengths encountered in these patterns are presented in Figure 3-9 of Reference (12) and 3-12 of Reference (13). The maximum rod strength permitted by the patterns is less than 0.01 Δk which is below strengths which could threaten the reactor coolant system. Above 10% power even single operator errors cannot result in out-of-sequence control rod worths which are sufficient to reach a peak fuel enthalphy content of 280 cal/gm; thus, requiring operation of the RWM or verification by a second licensed operator that the operator at the reactor console is following the rod program below 10% rated power is conservative.

A parametric analysis of the control rod drop accident was made in Reference (14), assuming the worst measured rod drop velocity and Technical Specification scram times, and the results indicate that a maximum in sequence rod worth of 1.25% Δk is acceptable.

The control rod system is designed to bring the reactor subcritical from a scram signal at a rate fast enough to prevent fuel damage. Figure III-1 of Amendment 69 to the FDSAR shows the control rod scram reactivity used in the transient analyses. Under these conditions, the thermal limits are never reached during the transients requiring control rod scram as presented in the FDSAR. The limiting power transient is that resulting from a turbine stop valve closure with failure of the turbine bypass system. Analysis of this transient show that the negative reactivity rates resulting from a flux scram with the average response of all operable drives in conformance with

be predictable to the equivalent of one percent in reactivity. Deviations beyond this magnitude would not be expected and would require thorough evaluation. One percent reactivity limit is considered safe since an insertion of this reactivity into the core would not lead to transients exceeding design conditions of the reactor system.

The scram reactivity function which results from a typical end-of-cycle control rod density was presented in Reference 15 and used in the bounding transient evaluations for equilibrium 7 x 7 fuel and 8 x 8 fuel reload cores. The effects of various off-design control rod patterns on the scram reactivity function were examined in Reference 16. The results indicated that the control rod density, not distribution, is the most significant parameter. Control rod densities as low as 3.5 percent were examined with essentially the same results as for the scram reactivity function used in Reference 15. Lower control rod densities would require scram bank worth predictions in comparison to the 3.5 percent control rod density predictions to support the contention that the effects of reduced control rod density maintain adequate margins to limits in the transient analyses. 25

References:

(1) FDSAR, Volume I, Section III - 5.3.1
(2) FDSAR, Volume II, Figure III - 5-11
(3) FDSAR, Volume I, Section VI-3
(4) FDSAR, Volume I, Section III - 5.2.1
(5) FDSAR, Volume I, Section VII-9
(6) FDSAR, Volume I, Section III - 5.2.2
(7) Licensing Application Amendment 11, Question II-3
(8) FDSAR, Volume I, Section III-5 and Volume II, Appendix B
(9) FDSAR, Volume I, Sections VII - 4.2.2 and VII - 4.3.1
(10) FDSAR, Volume I, Section VI-4
(11) License Application Amendment No. 55, Section 2
(12) Paone, C. J., Stirn, R.C., and Wooley, J. A., "Rod Drop Accident Analysis for Large Boiling Water Reactors," NEDO-10527, March 1972
(13) Paone, C. J., Stirn, R. C., and Baun, J. M., "Rod Drop Accident Analysis for Large Boiling Water Reactors, Addendum No. 2 Exposed Cores," NEDO-10527, Supplement 2, January 1973 25
(14) Oyster Creek Nuclear Generating Station, Docket No. 50-219, Amendment 74, "Rod Drop Accident Analysis," May 31, 1974
(15) Licensing Application Amendment No. 76, XN-74-43 (Revision 2) and XN-74-41 (Revision 2), dated January 31, 1975
(16) Oyster Creek Licensing Submittal, "Cycle 5 Reload and Loss-of-Coolant Accidnet Analysis Re-Evaluation", dated April 30 1975, Response 16G.

3.10 CORE LIMITS

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

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

Specification: A. Average Planar LHGR

During 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:

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

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
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 $\geq$ 1.73 for 8x8 fuel and $\geq$ 1.69 for 7x7 fuel. If at any time during steady state power operation it is determined by normal surveillance that the limiting value for MCPR is being exceeded action shall then be initiated to restore operation to within the prescribed limits. Surveillance and corresponding action shall continue until the prescribed limits are again being met. 25

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 $\pm$ 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) 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.

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: 25

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.

Power operation up to the fuel cladding integrity safety limit provides protection against a MCPR of 1.4 for 8 x 8 fuel and 1.37 for 7 x 7 fuel. The actual steady state operating power level provides margin to this limit by virtue of the overpower ratio of 1.236. The MCPR associated with this steady state power level is obtained by calculating the power which results in a boiling transition at those conditions. The ratio of this critical power to the steady state operating power is the Minimum Critical Power Ratio. This value is 1.73 for 8 x 8 fuel and 1.69 for 7 x 7. This provides in effect a margin to a boiling transition which is the product of the safety limit MCPR of 1.4 for 8 x 8 fuel and 1.37 for 7 x 7 fuel and the overpower ratio of 1.236.

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.

F. At specific power operating conditions, the actual control rod configuration will be compared with the expected configuration based upon appropriately corrected past data. This comparison shall be made every equivalent full power month. The initial rod inventory measurement performed when equilibrium conditions are established after a refueling or major core alteration will be used as base data for reactivity monitoring during subsequent power operation throughout the fuel cycle.

G. At power operating conditions, the actual control rod density will be compared with the 3.5 percent control rod density included in Specification 3.2.B.6. This comparison shall be made every equivalent full power month. 25

Basis: The core reactivity limitation (Specification 3.2.A) requires that core reactivity be limited such that the core could be made subcritical at any time during the operating cycle, with the strongest operable control rod fully withdrawn and all other operable rods fully inserted. Compliance with this requirement can be demonstrated conveniently only at the time of refueling. Therefore, the demonstration must be such that it will apply to the entire subsequent fuel cycle. The demonstration is performed with the reactor core in the cold, xenon-free condition and will show that the reactor is sub-critical at that time by at least R + 0.25% Δk with the highest worth operable control rod fully withdrawn.

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

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 least every 72 hours thereafter during power operation.

D. Minimum Critical Power Ratio (MCPR)

MCPR shall be checked every 72 hours during reactor operation and compared with the steady state value given in Specification 3.10.D. 25

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 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 to change significantly during steady state power operation so that 72 hours is an acceptable frequency for surveillance. 25

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 9 TO PROVISIONAL OPERATING LICENSE NO. DPR-16

(CHANGE NO. 25 TO THE TECHNICAL SPECIFICATIONS)

JERSEY CENTRAL POWER & LIGHT COMPANY

OYSTER CREEK NUCLEAR GENERATING STATION

Introduction

By application dated January 31, 1975, Jersey Central Power and Light Company (JCP&L) requested a license amendment authorizing operation of the Oyster Creek Nuclear Generating Station following reloading for core Cycle 5. According to JCP&L's application, approximately sixty-eight 8x8 reload fuel assemblies and thirty-six 7x7 reload fuel assemblies will replace an equal number of fuel assemblies in the core. The license amendment request included safety analyses for Cycle 5 and related Technical Specifications.

Supplemental information and responses to requests for additional information posed by the NRC staff were provided in JCP&L's letters dated March 25 and 29, 1975, April 24 and 30, 1975 and May 14 and 23, 1975.

By letter dated May 31, 1974 and revised by letter dated January 30, 1975, JCP&L requested a license amendment that would change the Technical Specifications to reduce the maximum allowable in-sequence control rod reactivity worth and would change the Technical Specifications to require a greater degree of operability of the rod worth minimizer and to change related procedural controls during reactor startup. Additional modifications to the proposed Technical Specifications relating to the rod worth minimizer were made based on discussions with JCP&L.

Our review of the JCP&L evaluation of the Oyster Creek Nuclear Generating Station ECCS performance is the subject of a separate safety evaluation Notice of the proposed action regarding the Cycle 5 reload, the maximum allowable in-sequence control rod worth, and the rod worth minimizer was published in the Federal Register on March 28, 1975 (40 FR 14123).

Evaluation

1.0 Fuel Design

The Oyster Creek Cycle 5 reload (reference 1) consists of approximately 36 Exxon Nuclear Company (Exxon) Type IIIF (7x7) fuel bundles, 64 Exxon Type VB fuel (8x8) bundles and four Exxon Type V (8x8) fuel bundles. The characteristics of the Type IIIF fuel were described in reference 2 and were approved by us for use in Cycle 4 (reference 3). Type V fuel

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characteristics were presented in Facility Change Request No. 6 (reference 4). The Type VB fuel described in the Cycle 5 reload submittal is the same as the Type V fuel except for: (1) a decrease in fuel enrichment and burnable poison content, and (2) a decrease in fuel pellet density. A comparison between the Exxon supplied Type IIIF (7x7), Type V (8x8), and Type VB (8x8) fuel assemblies are shown on Table 2. The smaller diameter 8x8 rods have a lower maximum linear heat generation rate, and a larger cladding thickness to diameter ratio which results in increased safety margins when compared to the 7x7 fuel assemblies. In particular, the maximum design linear power and maximum fuel temperature are substantially reduced with the 8x8 fuel design.

Fuel performance calculations that account for the effects of fuel densification, namely the potential for cladding collapse into axial gaps, the increase in the linear heat generation rate, the increase in the stored energy, and the increased probability of a local power spike resulting from axial gaps have been performed with the NRC staff approved version (reference 5) of the Exxon densification effects report (reference 6). The primary effects of densification on the fuel rod mechanical design are manifested in calculations on fuel-clad gap conductance and cladding collapse time. The approved analytical model incorporates time-dependent fuel densification, gap closure, and cladding creepdown for the calculation of gap conductance.

The approved Exxon clad collapse model (reference 6) has been used to calculate the collapse times for the 8x8 fuel assembly designs. The cladding collapse time calculated assuming the worst initial ovality and thinnest cladding acceptable under the fabrication specifications and taking no credit for fission gas release were in excess of the exposure lifetime of the fuel.

The control rod withdrawal error transient and fuel assembly loading error have been analyzed for the 7x7 and 8x8 fuel assemblies and neither incident results in a cladding strain above the 0.75% design basis limit or in fuel centerline melting for either type assembly.

JCP&L has provided a description of the in-reactor surveillance program proposed for the Exxon Type V 8x8 fuel assemblies. Included in this program are pre-irradiation, interim and post-irradiation measurements on ten fuel rods (five from each of two Type V assemblies), and a schedule for submitting inspection results to us. The interim examination will provide assurance that these assemblies are performing satisfactorily during the intended design lifetime and the post irradiation examinations will provide information in regard to margins of safety built into the generic 8x8 design. If these fuel assemblies do not perform as expected, the interim inspections will reveal any problems and give ample time for suitable action. In addition, Exxon has submitted a fuel performance report (reference 7) which includes results from inspections of the four Exxon Oyster Creek 7x7 lead fuel assemblies which have been in the reactor for two cycles and for the 148 Oyster Creek Type IIIE fuel reload assemblies which have been in the reactor for one cycle. Results from the "sipping" tests performed indicated there were no leaking assemblies.

Based on our review of the Oyster Creek Cycle 5 reload submittal, we conclude: (1) the fuel rod mechanical design provides acceptable engineering safety margins for normal operation, (2) the effects of fuel densification have been acceptably accounted for in the fuel design, (3) the 8x8 fuel assembly design, which has a maximum LHGR of 14.5 kw/ft, results in an increased margin in safety compared to the 7x7 fuel assembly design which has a maximum LHGR of 17.2 kw/ft, and (4) an adequate fuel surveillance program has been presented for the 8x8 fuel assemblies. We conclude that the fuel design for the Type IIIF (7x7), Type V (8x8), and Type VB (8x8) fuel assemblies is acceptable.

2.0 Nuclear Characteristics

The 104 Cycle 5 reload assemblies are arranged in the core so that the four fuel assemblies surrounding a control rod will contain only one new fuel assembly. The two rows and two columns of fuel bundles which intersect at the center of the reactor and a number of peripheral locations will not contain any new fuel. This loading scheme will ensure that a higher enrichment reload fuel assembly will be "paired" with three lower powered, exposed fuel assemblies. If the reactivity removed in the discharged fuel assemblies from the previous cycle is higher than expected or if localized power flattening is required, eight additional Type VB fuel assemblies will be loaded for Cycle 5. The reconstituted core for Cycle 5 will have quarter core symmetry.

The reload fuel assembly average enrichments and water to uranium dioxide volume ratio are similar to fuel assemblies manufactured by another vendor. However, the design details of the Oyster Creek reload fuel assemblies are different from those of the other vendor. For example, the number, enrichment and placement of fueled rods, and the use of four zirconium filled rods in the 8x8 Oyster Creek fuel assembly are notable differences. The nuclear characteristics of the reload fuel assemblies and the parameters of the reconstituted core for Oyster Creek Cycle 5 are presented in Tables III A-2, III B-1, and V B-1 of reference 1. Table V B-1 of reference 1 indicates that the values of the effective delayed neutron fraction, the neutron lifetime, the moderator void coefficient of reactivity, and the fuel Doppler coefficient of reactivity are nearly the same for fuel Cycle 5 as for Cycle 4. A list of important parameters for Type III-B and V-B reload fuel is given in Table 1.

The full power scram reactivity function for Cycle 5 is shown on Figure IV C-1 for the end of cycle rod pattern shown on Figure IV C-2 of reference 1. Also shown on Figure IV C-1 is a bounding curve which is used in the transient analysis. JCP&L's response to our request for additional information concerning the assumptions made in obtaining the scram functions used in Cycle 5 transient analyses indicates that the assumptions made in the calculation are conservative and that the bounding curve shown in Figure IV C-1 provides at least a 20% margin for off-design rod configurations in the crucial first two seconds after a scram has been initiated. Since the transient analyses were performed with a minimum control rod density of 3.4%, the Technical Specifications include a requirement which prohibits operation with a lower control rod density.

The shutdown margin of the reconstituted core meets the Technical Specification requirement that the core be at least 0.25% Δk subcritical in the most reactive operating state with the largest worth control rod fully withdrawn and all other control rods fully inserted. The shutdown margin is 0.01 Δk for the Cycle 5 core loading at the beginning of the cycle. Due to the burnout of the gadolinia in the reload fuel assemblies, the shutdown margin remains constant up to a cycle exposure of 1270 MWD/MTM and then increases throughout the remaining exposure of the cycle.

In response to a request for additional information on the liquid poison control system, JCP&L stated that the concentration of 600 ppm of boron for the Cycle 5 core has a reactivity worth of 0.1305 Δk/k. This boron reactivity worth will make the cold, xenon-free Cycle 5 core subcritical with a K_{eff} = 0.988. Therefore, the alternate shutdown requirement of the General Design Criteria is met.

The criterion for the storage of fuel for Oyster Creek is that the effective multiplication factor of the fuel as stored in the fuel storage pool must be 0.90. The K_{eff} of the Type IIIF fuel assembly in its most reactive state (zero exposure, gadolinia free) is calculated to be less than 0.88. The reactivity of a Type V B fuel assembly is nearly the same as for the Type IIIF fuel assembly. Therefore, the Cycle 5 reload fuel assemblies meet the fuel storage criterion of $K_{eff} \leq 0.90$.

Thus, based on our review of the information presented in the Oyster Creek Cycle 5 licensing submittal, we conclude that the nuclear characteristics and performance of the reconstituted core of Cycle 5 will not differ significantly from previous Oyster Creek cycles.

3.0 Fuel Assembly Loading Error

JCP&L's fuel assembly loading error analysis states that the calculated increase in Type V B assembly local power resulting from a 180° bundle rotation is 17%, which is less than the 21% calculated for a Type IIIF fuel assembly. A 180° assembly mis-orientation of a Type IIIF fuel assembly would result in a linear heat generation rate of 20 kW/ft at rated power (reference 8). As discussed in Section 1.0 of this evaluation, this loading error neither causes cladding strain to exceed the 0.75% design basis limit or produces fuel centerline melting. The interchange of a Type V B with a Type IIIF fuel assembly will not result in any limits being exceed since the 8x8 Type V B fuel assemblies are already in the more limiting core locations. Therefore, we conclude that the consequences of a fuel loading error are acceptable.

4.0 Minimum Critical Power Ratio Thermal Limit Criteria

4.1 Introduction

For the purpose of protecting the fuel cladding integrity, the power is maintained at an operating level such that the worst anticipated transient will lead to power and flow conditions providing adequate margin from the onset of "boiling transition"; this is assured when the critical power ratio (CPR) is no less that 1.40. The boiling transition correlation used by the applicant to calculate the CPR is the XN-2 correlation.

4.2 The XN-2 Correlation

This correlation and its application to the calculation of the critical average channel heat flux (effectively critical power) are described in Appendix B of reference 9, and in the errata given in reference 10. The procedure involves the comparison of the channel average heat flux with a critical value predicted by the XN-1 correlation (reference 11) divided by an axial shape corrector and the local radial peaking factor. When the two values are found to be equal at one point over the assembly length, boiling transition is assumed to occur, and the associated assembly power is the Critical Power, P_c. For a limiting CPR value of 1.40 the assembly power must not exceed a value of $P_c/1.40$.

We have reviewed the acceptability of this CPR value of 1.4 and the way in which the licensee has applied it to the Oyster Creek core analysis. The results of this review are discussed below.

4.3 Acceptability of a CPR of 1.40

To justify a CPR of 1.40, the licensee has referenced a GE report (reference 12) containing the results of over 1,000 boiling transition tests which adequately cover the operating range of the Oyster Creek core. In the analysis of this data, the licensee has found that XN-2 correlation overpredicts this data by a mean of 17.7% and that for a CPR of 1.40, there is a statistical confidence level in excess of 95% that the probability of boiling transition occurring on the hot rod is less than 5%. We have used the same data and independently arrived at the same conclusions.

Furthermore, the licensee has verified that a penalty factor of 3.3% in critical heat flux, justified in Appendix A of reference 9, for the four unfueled rods in the 8x8 fuel design, is accommodated within the above statistics.

We conclude that the XN-2 correlation may be used for the Oyster Creek core with a CPR limit line of 1.40 within the following ranges of parameters:

Mass flux: 0.3 to 1.1 x 10^6 lb/hr - ft^2
Pressure: 900 to 1250 psia
Inlet Subcooling: 10 to 55 Btu/lb
Local radial peaking factor: 1.32
Axial Peaking: inlet, cosine, and outlet

4.4 Application to the Oyster Creek Core Analysis

The transients have been analysed in terms of critical heat flux ratios (CHFR). Therefore, it has been necessary to relate these CHFR values to CPR values . The licensee has done this parametrically, e.g.: for several assumed power levels at which rod block occurs and for the corresponding power distributions, the licensee has evaluated MCPR and MCHFR at several steps in power above the assumed rod block power. The CHFR which corresponds to a CPR of 1.40 was found by interpolation. Thus, a CPR of 1.40 has a corresponding CHFR which differs according to the

type of transient. This method of assuring that the limiting CPR of 1.40 will not be exceeded during transients is acceptable to the staff.

In establishing the operating power to accommodate the most severe transient the licensee has used conservative values of the influencing parameters, namely peaking factors, flow, pressure and inlet subcooling.

In summary, the MCPR criterion and the methods of application to the Oyster Creek core are acceptable.

5.0 Transient Analysis

The design bases applied for abnormal transients are: (1) the system pressure does not exceed the ASME code limits of 100% of the vessel design pressure (1250 psig), (2) the core water level remains above the top of the active fuel, and (3) the margin to critical heat flux (CHF) in the core is sufficient to preclude fuel damage. Assuring that the CHF ratio does not go below 2.03 for 8x8 fuel and 1.80 for 7x7 fuel, provides sufficient margin to CHF. The linear heat generation rate (LHGR) and exposure limits to preclude fuel damage are discussed in the fuel design section.

The licensee presented bounding transient analyses by assuming separate equilibrium 7x7 and equilibrium 8x8 cores rather than using a mixed core loading of 7x7 and 8x8 fuel assemblies. The average thermal-hydraulic characteristics of a mixed core of 7x7 and 8x8 assemblies will be bounded by the average thermal-hydraulic characteristics of an all 7x7 fuel core and an all 8x8 fueled core. The hot channel MCHFR calculation can be affected by a mixed core if the hydraulic characteristics of the two assemblies are different. The core flow for a given pressure drop is approximately 1% less for the 8x8 assemblies. This core flow reduction is accounted for in the calculation of the minimum critical heat flux. The effect of this slight flow reduction on MCHFR is compensated by the fact that the 8x8 fuel assembly operates at lower heat fluxes than the 7x7 fuel assembly. The net result is higher operating MCHFR for the 8x8 assembly. This is also seen by the higher allowable peaking factors on the 8x8 fuel assembly.

To calculate CHF, the licensee presented a new correlation (XN-2) which was compared to data from experimental measurements of the critical power that causes boiling transition at conditions characteristic of boiling water reactors. This required establishing a revised core thermal-hydraulic limit for anticipated transients. The most limiting thermal-hydraulic transient (single rod withdrawal error transient) is used to determine the set-point for the neutron flux rod block that is necessary to insure that the power equivalent to a minimum critical power ratio (MCPR) is not violated. This MCPR is 1.40 for 8x8 fuel assemblies as determined by the XN-2 correlation. Since all transients were previously analyzed using the XN-1 critical heat flux correlation, it is necessary to determine a MCHFR which is equivalent to the MCPR determined by the XN-2

correlation. This equivalent value is used to demonstrate that limiting anticipated transients do not violate the established XN-2 MCPR's of 1.40 (8x8) or 1.37 (7x7).

The transition to a MCPR thermal-hydraulic limit involves Technical Specification changes for the Safety Limit, Limiting Safety System Settings, and Limiting Conditions of Operation to preclude violating the Safety Limit. The established XN-2 related MCHFR, corresponding to a MCPR of 1.4, is 2.03 for 8x8 fuel and for a MCPR of 1.37, the corresponding MCHFR is 1.80 for 7x7 fuel.

The licensee presented results of the limiting transients from the following categories: pressurization transients, core flow reductions, core flow increases, loss of feedwater, loss of feedwater heating and control rod withdrawal error transient. Of these transients the limiting MCHFR, assuming equilibrium of the non-mixed 7x7 and 8x8 fueled cores, results from one of the pressurization transients--a turbine trip without bypass. If this transient is initiated from the rated power of 1930 MWt, the MCHFR values fall below the limiting values. Consequently, it was determined that by reducing the operating power to 1820 MWt the MCHFR values fall above the limiting values and therefore are acceptable. More favorable power shapes would permit operation at higher power levels.

During the turbine trip without bypass from 1930 MWt the pressure at the safety valves peaks at 1201 psia and, at the core midplane it peaks at 1217 psia for the worst case. These values are below the lowest safety valve setpoint of 1227 psia. With the lower power levels imposed by the limiting MCHFR transient, we conclude that the analysis described above is conservative.

The limiting safety valve sizing transient for overpressure protection is a turbine trip with the following assumptions: (1) initial thermal power of 1930 MWt, (2) no credit for scram activations, (3) turbine stop valves close, and (4) no credit for solenoid-operated relief valves, bypass valves, and isolation condenser valves. For this transient, the peak pressure at the vessel bottom is 1354 psia for a 7x7 fueled core, and 1357 psia for an 8x8 fueled core which are below the ASME code limit of 1375 psia.

The limiting core flow increase transient, flow controller malfunction (maximum flow demand from low power) does not approach a limiting thermal condition because the surface heat flux does not increase above the operating steady-state value; therefore, MCHFR is not reduced below the initial steady-state value.

The limiting cold water addition transient, excess feedwater flow (at low power), results in an XN-2 MCHFR greater than that calculated for the turbine trip without bypass transient. A change in the steady-state operating power level at rated conditions will not influence the results of this transient. Hence, this transient is bounded conservatively by the constraints derived from the analysis of turbine trip without bypass.

Because the Cycle 5 operating power will be less than 1930 MWt as discussed previously, the loss of feedwater heater transient will result in a lower power level during the transient than previously analyzed. Hence, the results of the previous transient at an initial power of 1930 MWt is conservative.

The limiting reactor vessel water level transient is the loss of feedwater. The analysis showed a minimum level of 4.5 feet above the active core for the limiting 7x7 assembly when the transient is initiated from an initial steady-state power of 1930 MWt. Because the water level reduction was demonstrated to be greater when started from higher initial steady-state powers, therefore, the previous analysis is conservative when the transient is initiated from any power levels lower than 1930 MWt, required by the constraints discussed above.

The control rod withdrawal error transient was analyzed for the planned Cycle 5 core configuration. The starting conditions assumed are: (1) peak Cycle 5 reactivity, (2) a control rod pattern with no xenon or samarium present, and (3) full thermal power. It was shown that for the case in which a control rod is blocked when it is withdrawn 3.5 feet, when the resultant power shape is taken into account, the limiting power must be 1922 MWt to maintain the CPR margin of 1.40. Since the initial steady-state power must be no greater than 0.962 times this value to ensure this margin the operating power must not exceed 1848 MWt for the rod withdrawal error transient. Thus, given that the steady-state operating power does not exceed 1848 MWt, an APRM rod block setting of 1922 MWt at 100% flow is adequate to protect a MCFR of 1.40 for the limiting power shapes. More favorable power shapes would permit operation at higher power levels. The rod block setting is set forth in the Technical Specifications.

The transient analyses discussed above have demonstrated that the results are within the design bases considered applicable for abnormal operating transients.

6.0 Accident Analyses

6.1 Control Rod Drop Accident

The control rod drop accident for Oyster Creek Cycle 5 is presented in reference 1 and in response to our request for additional information. The accident reactivity shape and magnitude, the scram reactivity, and the Doppler coefficient of reactivity for Cycle 5 are compared with the analysis presented in reference 13. This comparison establishes the maximum allowable in-sequence rod worth as 0.0125 $\Delta k/k$ in order that the consequences of a control rod drop accident in Cycle 5 will not exceed the fuel stored energy design limit of 280 cal/gm. The analysis presented by JCP&L has been compared for a specific case with results obtained by our consultants at Brookhaven National Laboratory and the previous vendor and good agreement was shown. The method of analysis used by Oyster Creek closely parallels that developed by our consultants at BNL. The analysis of the control rod drop accident results in an enthalpy deposition greater than or equal to 170cal/gm for approximately 1.5% of the fuel rods in the Oyster Creek core. For a core with all

7x7 fuel this corresponds to a maximum of 450 perforated rods and for an 8x8 core a maximum of 550 perforated rods. Thus, in the event of a control rod drop accident, an Oyster Creek core with either all 7x7 or all 8x8 fuel would produce radiological consequences well below 10 CFR 100 guidelines.

Therefore, we conclude that the enthalpy deposition of a control rod drop accident for Cycle 5 will be less than the fuel stored-energy design limit of 280 cal/gm if the maximum allowed in-sequence control rod worth is equal to or less than 0.0125 $\Delta k/k$. We further conclude that the radiological consequences from assumed perforation of fuel rods with enthalpy deposition of 170 cal/gm, are well below the 10 CFR 100 guidelines.

6.2 Fuel Handling Accident

In the fuel handling accidents, the licensee calculated the number of failed fuel rods in the same manner as for the previous core loading. The total activity from an 8x8 assembly (60 fueled rods) is given as the same as for the 7x7 assemblies (48 fueled rods) except for slight changes in the relative amount of different constituents because of the presence of gadolinium in the Type VB bundles instead of boron in the curtains. Accordingly, under the worst postulated conditions, i.e., equivalent fission gas release from the fuel, the total activity release from a fuel handling accident with 8x8 assemblies would be equivalent to a fuel handling accident with 7x7 fuel assemblies. However, because of the lower fuel temperatures of 8x8 fuel, approximately 40% of the inventory of the fission gases in the 7x7 fuel rod gas spaces will be present in the fuel rod gas spaces of the 8x8 fuel rods. We therefore conclude that the consequences of the fuel handling accident with the 8x8 assemblies will not exceed consequences previously analyzed for the 7x7 assemblies.

6.3 Steam Line Break Accident

The analysis of the steam line break accident depends on the operating thermal-hydraulic parameters of the reactor and the overall factors affecting the consequences, such as coolant activity. The coolant activity is assumed to be the limiting value stated in the Technical Specifications. Reload fuel does not change these parameters; thus, the results of the Amendment 65 analysis, which we find acceptable, will not change.

7.0 Summary

Our evaluation supports the conclusion that the proposed safety limits, limiting safety system settings, limiting conditions for operation, and surveillance requirements were properly determined and have appropriately been incorporated in the proposed Technical Specifications.

8.0 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.

Dated MAY 2 4 1975

TABLE 1

PARAMETERS FOR CYCLE 5 RELOAD FUEL

Reload Fuel Type	III F (7 x 7)	V B (8 x 8)
Average Enrichment (wt % U-235)	2.63	2.50
Average Pellet Density (% TD)	94.5 and 95.0	93.5
Fueled Rods per Assemble	48	60
Fuel/% adolinia Rods per Assembly	4	4
Water to UO_2 Volume Ratio (cold)	2.51	2.61
Maximum Heating Rate (kW/ft)	17.2	14.5
Average Heating Rate (kW/ft)	5.8	4.6
Maximum Fuel Temperature (°F)	4120	3811

TABLE 2

COMPARISON OF THE CHARACTERISTICS OF TYPE VB, TYPE V, AND TYPE III F ASSEMBLIES

CHARACTERISTIC	TYPE IIIF (7x7)	TYPE V (8x8)	TYPE VB (8x8)
Maximum LHGR, kw/ft	17.2	14.5	14.5
UO_2 Fuel			
Enrichment, WT%			
Low	1.59	1.42	1.50
Medium	2.42	2.18	2.30
High	2.87	3.15	2.69
Bundle Average	2.63	2.65	2.50
Average Pellet Density % T.D.	94.5 & 95.0	94.5 & 95.0	93.5
Pellet Dish Volume, %	2.0	1.0	1.0
Zr-2 Cladding			
Ouside Diameter, in.	0.570	0.5015	0.5015
Thickness, Mils	35.5 & 45.5	0.360	0.360
Pellet to Cladding Gap, Mils	8 & 10	10	10
Fuel Bundle			
Number of Fuel Rods	48	60	60
Rod Pitch, inches	0.738	0.642	0.642
Water to UO_2 Volume Ratio (cold)	2.51	2.61	2.61

REFERENCES

1. Oyster Creek 1975 Reload for Cycle 5 Licensing Data Submittal, XN-74-52, Revision 3, January 22, 1975.

2. Oyster Creek Nuclear Generating Station, Facility Change Request No. 5, March 1, 1974.

3. Safety Evaluation by the Directorate of Licensing Supporting Amendment No. 4 to License DPR-16, (Change No. 20 to Appendix A of Technical Specifications).

4. Oyster Creek Nuclear Generating Station, Facility Change Request No. 6, April 16, 1974.

5. Technical Report on Densification of Exxon BWR Fuels, Supplement 1, U.S. Atomic Energy Commission, December 17, 1973.

6. Densification Effects on Exxon Nuclear Boiling Water Reactor Fuel, XN-174, November 1973.

7. Fuel Performance Report No. 2, XN-75-13, Exxon Nuclear Company, February 1975.

8. Oyster Creek Nuclear Generating Station - Facility Change Request No. 4, January 18, 1974.

9. Oyster Creek Amendment No. 76 (Supplement No. 3) and Technical Specification Change Request No. 37, May 14, 1975.

10. Letter from I. R. Finfrock to A. Giambusso dated May 23, 1975, subject: Errata.

11. Report JN-72-18, proprietary.

12. Report NEDE-10958, proprietary, referenced in a letter dated May 23, 1975, from I. R. Finfrock to A. Giambusso, subject: Supporting Information.

13. Oyster Creek Amendment No. 74 and Technical Specification Change Request No. 27, May 31, 1974.

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. 9 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 authorizes operation with 8x8 fuel assemblies and changes to the Technical Specifications in order to: (1) reduce the maximum allowable in sequence control rod reactivity worth, (2) require a greater degree of operability of the rod worth minimizer, and (3) change related procedural controls during reactor startup.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Notice of Proposed Issuance of Amendment to Provisional Operating License in connection with this action was published in the FEDERAL REGISTER on March 28, 1975 (40 F.R.14123). 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 May 31, 1974 and January 31, 1975, and supplements dated March 25 and 29, April 24 and 30, and May 14 and 23, 1975, (2) Amendment No. 9 to License No. DPR-16, with Change No. 25 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.

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 Reactor Licensing.

Dated at Bethesda, Maryland this MAY 24 1975

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

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