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. 36
License No. DPR-16

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

   A. The application for amendment by Jersey Central Power & Light Company (the licensee) dated May 19, 1979, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   B. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 36 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 30, 1979

ATTACHMENT TO LICENSE AMENDMENT NO. 36

PROVISIONAL OPERATING LICENSE NO. DPR-16

DOCKET NO. 50-219

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain vertical lines indicating the areas of change.

REMOVE	INSERT
2.1-2	2.1-2
--	2.1-2a
2.1-4	2.1-4
--	2.1-4a

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D. During all modes of reactor operation 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.

F. During all modes of operation except when the reactor head is off and the reactor is flooded to a level above the main steam nozzles, at least two (2) recirculation loop suction valves and their associated discharge valves will be in the full open position.

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 setpoint. 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.

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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.

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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 trubine 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.

If reactor water level should drop below the top of the active fuel, 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 presently be monitored is 4'-8" above the top of the active fuel. Although the lowest reactor water level limit which ensures adequate core cooling is the top of the active fuel, the safety limit has been established at 4'-8" to provide a point which can be monitored.

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Specification F assures that an adequate flow path exists from the annular space, between the pressure vessel wall and the core shroud, to the core region. This provides for good communication between these areas, thus assuring that reactor water level instrument readings are truly indicative of the water level in the core region.