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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 36 TO LICENSE NO. DPR-16

JERSEY CENTRAL POWER & LIGHT COMPANY

OYSTER CREEK NUCLEAR GENERATING STATION

DOCKET NO. 50-219

Introduction

By letter dated May 19, 1979, Jersey Central Power & Light Company (the licensee) requested an amendment to the Technical Specifications of License No. DPR-16 for the Oyster Creek Nuclear Generating Station. The amendment would revise the Technical Specifications to extend the applicability of the minimum water level safety limit to all modes of operation, and to add a new safety limit to require that two recirculation loops remain open during all modes of operation except with the reactor vessel head removed.

Discussion

On May 2, 1979, during the performance of the isolation condenser automatic actuation surveillance test a false reactor high pressure scram occurred at the Oyster Creek Nuclear Generating Station. Subsequenctly, a turbine trip occurred on low load. This initiates an automatic transfer of power to the startup transformers. Startup transformer SA provides the power for the A feedwater train, and startup transformer SB provides power for the B & C feedwater pumps. However, SB was out of service for maintenance so when the main turbine generator tripped the power supply to the B & C feedwater pumps was lost. The A feedwater pump tripped because of the hydraulic transient caused by the loss of the B & C feedwater pumps. Therefore all three feedwater pumps were tripped.

During the loss of feedwater transient all five of the recirculation loop pump discharge valves were closed and all of the two inch bypass lines were open. These five bypass lines did not allow a large enough flow of water from the outside of the core region, the annulus, to the core region. As a result, the water was boiling away in the core region faster than it was being returned through the bypass lines and the water level above the core decreased below the low-low-low level alarm. When one of the recirculation loop pump discharge valves was reopened the water flow from the annulus to the core region was large enough to compensate for the water boiling off in the core region so the water level increased above the low-low-low level alarm.

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Existing Technical Specification 2.1.D defines the low-low-low level as a safety limit when the reactor is in the shutdown condition. Even though the reactor mode switch was not in the shutdown mode, the licensee and the NRC have been treating the May 2, 1979 event as if a safety limit had been violated. Therefore, the reactor was placed in the cold shutdown condition and the licensee and the NRC performed a thorough evaluation of the minimum water level that occurred during the event to determine if any fuel damage had occurred. In addition, the NRC conducted an evaluation of the follow up actions proposed by the licensee to prevent recurrence. This license amendment, is one part of that follow up action.

The NRC evaluation of the event, the condition of the core, and all of the corrective measures taken to prevent recurrence is being described in a separate Safety Evaluation Report. This evaluation is related only to the proposed license amendment.

Evaluation

As a result of the analysis of the May 2, 1979 event it was recognized by the licensee and the NRC staff that the low-low-low water level safety limit should be applied and clearly defined to include all modes of operation when the reactor vessel contains fuel. The basis for the current technical specifications limit was to assure adequate margin for removing decay heat from the fuel during periods when the reactor is shutdown and corresponds to the lowest reactor vessel water level that can be monitored. The basis for the safety limit during operation is to assure adequate margin of water above the top of the core to prevent core uncovery during anticipated transients. It is considered prudent to have a measurable water level limit for the safety limit for all modes of operation. Therefore, the licensee has requested that the technical specifications be modified to make the low-low-low water level (4 feet 8 inches above the top of the active fuel zone) a safety limit applicable to all modes of operation including transient conditions.

This change more clearly defines the safety limit for reactor vessel water level for all modes of operation, the low-low-low water level limit is not changed, therefore, the proposed change is acceptable.

The licensee has also proposed to add a safety limit as section 2.1.F which requires that during all modes of operation except when the reactor head is removed and the reactor is flooded to a level above the mainsteam nozzles at least two (2) recirculation loop pump suction valves and their associated discharge valves will be in the full open position. This will assure that at all appropriate times the water in the core and in the annulus will be in hydraulic communication to preclude recurrence of the May 2, 1979 event at Oyster Creek resulting from different levels in these regions.

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The effects of discharge valve position on steady-state water level in the core region have been evaluated from a hydraulic analysis of the recirculation lines. This analysis modeled the recirculation line geometry with standard fluid mechanic methods. This modeled the geometric pressure loss coefficients which includes a factor for the fixed rotor recirculation pump.

The pressure loss coefficient for the recirculation pump has been established from in situ tests. The other pressure loss coefficients are from standard methods and are adequate. The analysis assumed differential driving heads between annulus and core regions which are within the range of values assumed for overall analyses. These methods were utilized to calculate the natural circulation flow through one recirculation loop.

The distribution of coolant inventory (between annulus and core) has been accounted for based on no forced recirculation flow (due to a reactor coolant pump trip on low-low water level) and a maximum of one unisolated recirculation loop. The above conditions will result in the most adverse distribution of coolant inventory within the reactor vessel.

Based on our evaluation we have concluded that the proposed Technical Specification will assure continual hydraulic communication between the annulus and the core during all modes of operation including transients, and therefore, is acceptable.

Environmental Considerations

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

Conclusion

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

Date: May 30, 1979

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

May 30, 1979

Docket No. 50-219

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

Dear Mr. Finfrock:

The Commission has issued the enclosed Amendment No. 36 to Provisional Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station. This amendment consists of changes to the Technical Specifications in response to your application dated May 19, 1979.

The amendment modifies Section 2.1.D to extend the applicability of the minimum water level safety limit to all modes of operation, and add a new safety limit in Section 2.1.F to require that two recirculation loops remain open during all modes of operation except with the reactor vessel head removed.

Copies of our related Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

Dennis L. Ziemann

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

Enclosures:
1. Amendment No. 36 to DPR-16
2. Safety Evaluation
3. Notice of Issuance

cc w/enclosures:
See next page

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cc

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

JERSEY CENTRAL POWER & LIGHT COMPANY

DOCKET NO. 50-219

OYSTER CREEK NUCLEAR GENERATING STATION, UNIT NO. 1

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 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 regulations 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.

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.

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.

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.

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UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-219

JERSEY CENTRAL POWER & LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO PROVISIONAL OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 36 to Provisional Operating License No. DPR-16, issued to Jersey Central Power & Light Company (the licensee), which revised the Technical Specifications for operation of the Oyster Creek Nuclear Generating Station (the facility) located in Ocean County, New Jersey. The amendment is effective as of its date of issuance.

The amendment modifies Section 2.1.D to extend the applicability of the minimum water level safety limit to all modes of operation, and add a new safety limit in Section 2.1.F to require that two recirculation loops remain open during all modes of operation except with the reactor vessel head removed. In separate actions relating to this facility, the Commission is: (1) Amending License No. DPR-16 to allow operation with the more restrictive Maximum Average Planar Linear Heat Generation Rate limits authorized by Amendment No. 30, dated March 14, 1978, extended to encompass higher exposure fuel in the reactor; and (2) Authorizing resumption of operation after a May 2, 1979, transient event involving license safety limits.

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

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

For further details with respect to this action, see (1) the application for amendment dated May 19, 1979, (2) Amendment No. 36 to License No. DPR-16, and (3) the Commission's related Safety Evaluation. All of these items and the Commission's separate actions described above are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C., and at the Ocean County Library, Brick Township Branch, 401 Chambers Bridge Road, Brick Town, New Jersey 08723. A copy of items (2) and (3) in addition to the separate actions may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 30th day of May, 1979.

FOR THE NUCLEAR REGULATORY COMMISSION

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