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OCT 6 1975

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 requested the Office of the Federal Register to publish the enclosed Notice of Proposed Issuance of an Amendment to Provisional Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station. The proposed amendment includes a change to the Technical Specifications as proposed by our letter of July 16, 1975 and subsequently modified by a few mutually acceptable changes. Your staff has indicated that the comments in your letter of August 8, 1975, are now resolved and that the Technical Specifications, as modified, are acceptable.

This amendment would incorporate: (1) water temperature limits during any testing which adds heat to the suppression pool, (2) suppression pool water temperature limits requiring manual scram of the reactor, (3) suppression pool water temperature limits requiring reactor pressure vessel depressurization, (4) surveillance requirements to monitor water temperatures during operations which add heat to the suppression pool and (5) external visual examinations of the suppression chambers following operations in which the pool temperatures exceed 160°F.

In addition to the limits on the temperature of the suppression chamber pool water, your operating procedures should define the operator action to be taken in the event a relief valve inadvertently opens or sticks open. This action would include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling heat exchangers, (3) initiate reactor shutdown, and (4) if other relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open relief valve to assure mixing and uniformity of energy insertion to the pool.

Copies of the proposed amendment, the related Safety Evaluation, and the Federal Register Notice are enclosed.

Sincerely,

/S/

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

Concur

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Jersey Central Power & Light Company

cc: w/enclosures

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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

PROPOSED AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No.
License No. DPR-16

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

   A. 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

   B. 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:

   "(2) 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. __".

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

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

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

Attachment:
Change No. __ to the
Technical Specifications

Date of Issuance:

PROPOSED CHANGES TO

TECHNICAL SPECIFICATIONS

PROVISIONAL OPERATING LICENSE NO. DPR-16

DOCKET NO. 50-219

The proposed changes to the Technical Specifications are shown on the attached pages and are identified by a vertical line in the margin.

## 3.5 CONTAINMENT

Applicability: Applies to the operating status of the primary and secondary containment systems.

Objective: To assure the integrity of the primary and secondary containment systems.

Specifications: A. Primary Containment

1. At any time that the nuclear system is pressurized above atmospheric or work is being done which has the potential to drain the vessel, the suppression pool water volume and temperature shall be maintained within the following limits.

   a. Maximum water volume - 92,000 $ft^3$
   b. Minimum water volume - 82,000 $ft^3$
   c. Maximum water temperature

      (1) During normal power operation - 95°F

      (2) During testing which adds heat to the suppression pool, the water temperature shall not exceed 10F above the normal power operation limit specified in (1) above. In connection with such testing, the pool temperature must be reduced to below the normal power operation limit specified in (1) above within 24 hours.

      (3) The reactor shall be scrammed from any operating condition if the pool temperature reaches 110F. Power operation shall not be resumed until the pool temperature is reduced below the normal power operation limit specified in (1) above.

      (4) During reactor isolation conditions, the reactor pressure vessel shall be depressurized to less than 180 psig at normal cooldown rates if the pool temperature reaches 120F.

2. Primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel except while performing low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5 MWt.

3. Reactor Building to Suppression Chamber Vacuum Breaker System

   a. Except as specified in Specification 3.5.A.3.b below, two reactor building to suppression chamber vacuum breakers in each line shall be operable at all times when primary containment integrity is required. The set point of the differential pressure instrumentation which actuates the air-operated vacuum breakers shall not exceed 0.5 psid. The vacuum breakers shall move from closed to fully open when subjected to a force equivalent to not greater than 0.5 psid acting on the vacuum breaker disc.

   b. From the time that one of the reactor building to suppression chamber vacuum breakers is made or found to be inoperable, the vacuum breaker shall be locked closed and reactor operation is permissible only during the succeeding seven days unless such vacuum breaker is made operable sooner, provided that the procedure does not violate primary containment integrity.

   c. If the limits of Specification 3.5.A.3.a are exceeded, reactor shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

4. Pressure Suppression Chamber - Drywell Vacuum Breakers

   a. When primary containment is required, all suppression chamber - drywell vacuum breakers shall be operable except during testing and as stated in Specification 3.5.A.4.b and c, below. Suppression chamber - drywell vacuum breakers shall be considered operable if:

      (1) The valve is demonstrated to open from closed to fully open with the applied force at all valve positions not exceeding that equivalent to 0.5 psi acting on the suppression chamber face of the valve disk.

      (2) The valve disk will close by gravity to within not greater than 0.10 inch of any point on the seal surface of the disk when released after being opened by remote or manual means.

      (3) The position alarm system will annunciate in the control room if the valve is open more than 0.10 inch at any point along the seal surface of the disk.

b. Two of the fourteen suppression chamber - drywell vacuum breakers may be inoperable provided that they are secured in the closed position.

c. One position alarm circuit for each operable vacuum breaker may be inoperable for up to 15 days provided that each operable suppression chamber - drywell vacuum breaker with one defective alarm circuit is physically verified to be closed immediately and daily during this period.

5. After completion of the startup test program and demonstration of plant electrical output, the primary containment atmosphere shall be reduced to less than 5.0% $O_2$ with nitrogen gas within 24 hours after the reactor mode selector switch is placed in the run mode. Primary containment deinerting may commence 24 hours prior to a scheduled shutdown.

6. If specifications 3.5.A.1.a, b, c(1) and 3.5.A.2 through 3.5.A.5 cannot be met, reactor shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

B. Secondary Containment

1. Secondary containment integrity shall be maintained at all times unless all of the following conditions are met.

   a. The reactor is subcritical and Specification 3.2.A is met.

   b. The reactor is in the cold shutdown condition.

   c. The reactor vessel head or the drywell head are in place.

   d. No work is being performed on the reactor or its connected systems in the reactor building.

   e. No operations are being performed in, above, or around the spent fuel storage pool that could cause release of radioactive materials.

2. The standby gas treatment system shall be operable at all times when secondary containment integrity is required except as specified by Specification 3.5.B.3.

3. One standby gas treatment filter circuit may be inoperable for 7 days, when standby gas treatment system operability is required, except during reactor startup, provided the remaining filter circuit is proved operable daily.

4. If Specifications 3.5.B.2 and 3.5.B.3 are not met, reactor shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours and the conditions of Specification 3.5.B.1 shall be met.

Bases:

Specifications are placed on the operating status of the containment systems to assure their availability to control the release of any radioactive material from irradiated fuel in the event of an accident condition. The primary containment system (1) provides a barrier against uncontrolled release of fission products to the environs in the event of a break in the reactor coolant systems.

Whenever the reactor coolant water temperature is above 212°F, failure of the reactor coolant system would cause rapid expulsion of the coolant from the reactor with an associated pressure rise in the primary containment. Primary containment is required, therefore, to contain the thermal energy of the expelled coolant and fission products which could be released from any fuel failures resulting from the accident. If the reactor coolant is not above 212°F, there would be no pressure rise in the containment. In addition, the coolant cannot be expelled at a rate which could cause fuel failure to occur before the core spray system restores cooling to the core. Primary containment is not needed while performing low power physics tests since the rod worth minimizer would limit the worst case rod drop accident to 1.5%$\Delta$k. This amount of reactivity addition is insufficient to cause fuel damage.

The absorption chamber water volume provides the heat sink for the reactor coolant system energy released following the loss-of-coolant accident. The core spray pumps and containment spray pumps are located in the corner rooms and due to their proximity to the torus, the ambient temperature in those rooms could rise during the design basis accident. Calculations[7] made, assuming an initial torus water temperature of 100°F and a minimum water volume of 82,000 $ft^3$, indicate that the corner room ambient temperature would not exceed the core spray and containment spray pump motor operating temperature limits, and, therefore, would not adversely affect the long term core cooling capability. The maximum water volume limit allows for an operating range without significantly affecting the accident analyses with respect to free air volume in the absorption chamber. For example, the containment capability[8] with a maximum water volume of 92,000 $ft^3$ is reduced by not more than 3.5% metal-water reaction below the capability with 82,000 $ft^3$.

Experimental data indicates that excessive steam condensing loads can be avoided if the peak temperature of the suppression pool is maintained below 160°F during any period of relief valve operation with sonic conditions at the discharge exit. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high suppression chamber loadings.

The purpose of the vacuum relief valves is to equalize the pressure between the drywell and suppression chamber and suppression chamber and reactor building so that the containment external design pressure limits are not exceeded.

The vacuum relief system from the reactor building to the pressure suppression chamber consists of two 100% vacuum relief breaker subsystems (2 parallel sets of 2 valves in series). Operation of either subsystem will maintain the containment external pressure less than the external design pressure; the external design pressure of the drywell is 2 psi; the external design pressure of the suppression chamber is 1 psi (FDSAR Amendment 15, Section II).

P. Suppression Chamber Surveillance

1. At least once per day the suppression chamber water level and temperature and pressure suppression system pressure shall be checked.

2. A visual inspection of the suppression chamber interior, including water line regions, shall be made at each major refueling outage.

3. Whenever heat from relief valve operation is being added to the suppression pool, the pool temperature shall be continually monitored and also observed until the heat addition is terminated.

4. Whenever operation of a relief valve is indicated and the suppression pool temperature reaches 160F or above while the reactor primary coolant system pressure is greater than 180 psig, an external visual examination of the suppression chamber shall be made before resuming normal power operation.

**Basis:**

The primary containment preoperational test pressures are based upon the calculated primary containment pressure response in the event of a loss-of-coolant accident. The peak drywell pressure would be 38 psig which would rapidly reduce to 20 psig within 100 seconds following the pipe break. The total time the drywell pressure would be above 35 psig is calculated to be about 7 seconds. Following the pipe break absorption chamber pressure rises to 20 psig within 8 seconds, equalizes with drywell pressure at 25 psig within 60 seconds and thereafter rapidly decays with the drywell pressure decay. (1)

The design pressures of the drywell and absorption chamber are 62 psig and 35 psig, respectively. (2) The design leak rate is 0.5%/day at a pressure of 35 psig. As pointed out above, the pressure response of the drywell and absorption chamber following an accident would be the same after about 60 seconds. Based on the calculated primary containment pressure response discussed above and the absorption chamber design pressure, primary containment prepperational test pressures were chosen. Also, based on the primary containment pressure response and the fact that the drywell and absorption chamber function as a unit, the primary containment will be tested as a unit rather than testing the individual components separately.

The design basis loss-of-coolant accident was evaluated at the primary containment maximum allowable accident leak rate of 1. 0%/day at 35 psig. The analysis showed that with this leak rate and a standby gas treatment system filter efficiency of 90 percent for halogens, 95% for particulates, and assuming the fission product release fractions stated in TID-14844, the maximum total whole body passing cloud dose is about 10 rem and the maximum total

After the containment oxygen concentration has been reduced to meet the specification initially, the containment atmosphere is maintained above atmospheric pressure by the primary containment inerting system. This system supplies nitrogen makeup to the containment so that the very slight leakage from the containment is replaced by nitrogen, further reducing the oxygen concentration. In addition, the oxygen concentration is continuously recorded and high oxygen concnetration is annunciated. Therefore, a weekly check of oxygen concentration is adequate. This system also provides capability for determining if there is gross leakage from the containment.

The drywell exterior was coated with Firebar D prior to concrete pouring during construction. The Firebar D separated the drywell steel plate from the concrete. After installation, the drywell liner was heated and expanded to compress the Firebar D to supply a gap between the steel drywell and the concrete. The gap prevents contact of the drywell wall with the concrete which might cause excessive local stresses during drywell expansion in a loss-of-coolant accident. The surveillance program is being conducted to demonstrate that the Firebar D will maintain its integrity and not deteriorate throughout plant life. The surveillance frequency is adequate to detect any deterioration tendency of the material.[8]

The operability of the instrument line flow check valves are demonstrated to assure isolation capability for excess flow and to assure the operability of the instrument sensor when required.

Because of the large volume and thermal capacity of the suppression pool, the volume and temperature normally changes very slowly and monitoring these parameters daily is sufficient to establish any temperature trends. By requiring the suppression pool temperature to be continually monitored and also observed during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken. The requirement for an external visual examination following any event where potentially high loadings could occur provides assurance that no significant damage was encountered. Particular attention should be focused on structural discontinuities in the vicinity of the relief valve discharge since these are expected to be the points of highest stress.

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

# SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION SUPPORTING PROPOSED AMENDMENT TO PROVISIONAL OPERATING LICENSE DPR-16 AND PROPOSED CHANGES TO TECHNICAL SPECIFICATIONS SUPPRESSION POOL WATER TEMPERATURE LIMITS JERSEY CENTRAL POWER & LIGHT COMPANY OYSTER CREEK NUCLEAR GENERATING STATION DOCKET NO. 50-219

## Introduction

By letter dated February 15, 1975 to Jersey Central Power & Light Company, the Nuclear Regulatory Commission (NRC) requested that the licensee among other things, develop operating procedures and proposed changes to the Technical Specifications to preclude reaching elevated temperatures of the torus pool water and to provide for inspection of the torus as appropriate to identify any damage in the event of an extended relief valve operation. By letter dated April 1, 1975 Jersey Central submitted a response which stated that the present Technical Specifications provide adequate limits for the suppression chamber water temperature, thus the licensee proposed no change to the Technical Specifications. This response from the licensee was found unacceptable; and, as a result, the NRC staff prepared appropriate technical specification changes to revise the suppression pool water temperature limits for Oyster Creek Nuclear Generating Station. By letter dated July 16, 1975, the NRC staff advised the licensee of its intent to initiate steps to issue these technical specification changes unless the licensee objected in writing. By letter dated August 8, 1975, the licensee provided comments on the technical specification changes proposed by the NRC staff. Subsequent discussions between the NRC staff and the licensee resulted in a few mutually acceptable changes to the Technical Specifications proposed by the NRC staff. The licensee stated it would accept the proposed Technical Specifications with these changes.

## Discussion

Oyster Creek is a boiling water reactor (BWR) which is housed in a Mark I primary containment. The Mark I primary containment is a pressure suppression type of primary containment that consists of a drywell and a suppression chamber (also referred to as the torus). The suppression chamber, or torus, contains a pool of water and is designed to suppress the pressure during a postulated loss-of-coolant accident (LOCA) by condensing the steam released from the reactor primary system. The reactor system energy released by relief valve operation during operating transients also is released into the pool of water in the torus.

Experiences at various BWR plants with Mark I containments have shown that damage to the torus structure can occur from two phenomena associated with relief valve operations. Damage can result from the forces exerted on the structure when, on first opening the relief valves, steam and the air within the vent are discharged into the torus water. This phenomenon is referred to as steam vent clearing. The second source of potential structural damage stems from the vibrations which accompany extended relief valve discharge into the torus water if the pool water is at elevated temperatures. This effect is known as the steam quenching vibration phenomenon.

A. Steam Vent Clearing Phenomenon

With regard to the steam vent clearing phenomenon, we are actively reviewing this generic problem and in our letter dated February 15, 1975 we also requested the licensee to provide information to demonstrate that the torus structure of the primary containment will maintain its integrity throughout the anticipated life of the facility. In its response dated April 1, 1975 the licensee stated that it was investigating this matter and the results of the investigation would be submitted to us on a schedule consistent with the timing which we proposed for licensee repsonse. Because of the apparent slow progression of the material fatigue associated with the steam vent clearing phenomenon, we have concluded that there is no immediate potential hazard resulting from this type of phenomenon; nevertheless, surveillance and review action on this matter by the NRC staff will continue in due course during this year.

B. Steam Quenching Vibration Phenomenon

The steam quenching vibration phenomenon became a concern as a result of occurrences at two European reactors. With torus pool water temperatures increased in excess of 170F due to prolonged steam quenching from relief valve operation, hydrodynamic fluid vibrations occurred with subsequent moderate to high relief valve flow rates. These fluid vibrations produced large dynamic loads on the torus structure and extensive damage to torus internal structures. If allowed to continue, the dynamic loads could have resulted in structural damage to the torus itself due to material fatigue. Thus, the reported occurrences of the steam quenching vibration phenomenon at the two European reactors indicate that actual or incipient failure of the torus can occur from such an event.

Such failure would be expected to involve cracking of the torus wall and loss of containment integrity. Moreover, if a LOCA occurred simultaneously with or after such an event, the consequences could be excessive radiological doses to the public. In comparison with the steam vent clearing phenomenon, the potential risk associated with the steam quenching vibration phenomenon (1) reflects the fact that a generally smaller safety margin1/ exists between the present license requirements on suppression pool temperature limits and the point at which damage could begin and (2) is more immediate.

## Evaluation

The existing Technical Specifications for Oyster Creek limit the torus pool temperature to 100F. This temperature limit has been reduced to 95F to provide 5F temperature difference between a scram requirement discussed below and provisions for performing necessary surveillance. The temperature of 95F assures that the pool water has the capability to perform as a constantly available heat-sink with a reasonable operating temperature that can be maintained by use of heat exchangers whose secondary cooling water (the service cooling water) is expected to remain below 95F. While this 95F limit provides normal operating flexibility, short-term temperatures permitted by operating procedures exceed the normal power operating temperature limit, but accommodates the heat release resulting from abnormal operation, such as relief valve malfunction, while still maintaining the required heat-sink (absorption) capacity of the pool water needed for the postulated LOCA conditions. However, in view of the potential risk associated with the steam quenching vibration phenomenon, it is necessary to modify the temperature limits now in the license Technical Specifications. This action was, as discussed in our February 15, 1975 letter, first suggested by General Electric Company (GE) who had earlier informed us of the steam quenching vibration occurrences at a meeting on November 1, 1974 and provided related information by letters to us dated November 7, and December 20, 1974. The December 20 letter stated that GE had informed all of its customers with operating BWR facilities and Mark I containments of the phenomenon and included in these communications GE's recommended interim operating temperature limits and proposed operating procedures to minimize the probability of encountering the damaging regime of the steam quenching vibration phenomenon.

Our implementation of the GE recommended procedures and temperature limits via changes in the Technical Specifications are evaluated in the following paragraphs:

1/ The difference, in pool water temperature, between the license limit(s) and the temperature at which structural damage might occur is the safety margin available to protect against the effects of the phenomenon discussed.

a. The new short-term limit applicable to all conditions requires that the reactor be scrammed if the torus pool water temperature reaches 110F. This requirement to scram at 110F provides additional assurance that the torus temperature will remain below the 170F temperature related to potential damage to that torus.

b. For specific requirements associated with surveillance testing, i.e., testing of relief valves, the water temperature shall not exceed 10F above the normal power operation limit. This new limit during surveillance testing of relief valves provides additional operating flexibility while still maintaining a maximum heat-sink capacity. The current limits in the Technical Specifications make no provision for these requirements.

c. For reactor isolation conditions, the new temperature limit is 120F, above which temperature the reactor vessel is to be depressurized. This new limit of 120F assures pool capacity for absorption of heat released to the torus while avoiding undesirable reactor vessel cooldown transients. Upon reaching 120F, the reactor is placed in the cold, shutdown condition at the fastest rate consistent with the technical specifications on reactor pressure vessel cooldown rates.

Conclusion

We have concluded, based on the consideration discussed above that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: OCT 6 1975

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-219

JERSEY CENTRAL POWER & LIGHT COMPANY

NOTICE OF PROPOSED ISSUANCE OF AMENDMENT

TO PROVISIONAL OPERATING LICENSE

The Nuclear Regulatory Commission (the Commission) is considering issuance of an amendment to Provisional Operating License No. DPR-16 issued to Jersey Central Power & Light Company (the licensee), for operation of the Oyster Creek Nuclear Generating Station, located in Ocean County, New Jersey.

The amendment would modify the provisions in the Technical Specifications relating to temperature limits for the pressure suppression pool water.

Prior to issuance of the proposed license amendment, the Commission will have made the findings required by the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations.

By 11/14/75 the licensee may file a request for a hearing and any person whose interest may be affected by this proceeding may file a request for a hearing in the form of a petition for leave to intervene with respect to the issuance of the amendment to the subject provisional operating license. Petitions for leave to intervene must be filed under oath or affirmation in accordance with the provisions of Section 2.714 of 10 CFR Part 2 of the Commission's regulations. A petition for leave to intervene must set forth the interest of the petitioner in the proceeding, how that interest may be affected by the results of the proceeding, and the petitioner's contentions with respect to the proposed licensing action. Such petitions must be filed in accordance with the provisions of this FEDERAL REGISTER notice and Section 2.714, and must be filed with the Secretary of the Commission, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Docketing and Service Section, by the above date. A copy of the petition and/or request for a hearing should be sent to the Executive Legal Director, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, and to George F. Trowbridge, Esq., Shaw, Pittman, Potts, Trowbridge & Madden, 910 - 17th Street, N. W., Washington, D. C. 20016, the attorney for the licensee.

A petition for leave to intervene must be accompanied by a supporting affidavit which identifies the specific aspect or aspects of the proceeding as to which intervention is desired and specifies with particularity the facts on which the petitioner relies as to both his interest and his contentions with regard to each aspect on which intervention is requested. Petitions stating contentions relating only to matters outside the Commission's jurisdiction will be denied.

All petitions will be acted upon by the Commission or licensing board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel. Timely petitions will be considered to determine whether a hearing should be noticed or another appropriate order issued regarding the disposition of the petitions.

In the event that a hearing is held and a person is permitted to intervene, he becomes a party to the proceeding and has a right to participate fully in the conduct of the hearing. For example, he may present evidence and examine and cross-examine witnesses.

For further details with respect to this action, see the letter from K. Goller to I. R. Finfrock, Jr. dated July 16, 1975 and the letter from I. R. Finfrock, Jr. dated August 8, 1975, which are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Ocean County Library, 15 Hooper Avenue, Toms River, New Jersey 08753. The proposed license amendment and the Safety Evaluation, may be inspected at the above locations and a copy 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 6th day of October, 1975.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

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

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**UNITED STATES NUCLEAR REGULATORY COMMISSION**

**DOCKET NO. 50-219**

**JERSEY CENTRAL LPOWER AND LIGHT COMPANY**

**PROPOSED ISSUANCE OF AMENDMENT TO**

**PROVISIONAL OPERATING LICENSE**

**(CORRECTION)**

In FR Document 75-27663 appearing at page 48407 in the Federal Register of Wednesday, October 15, 1975, correction is hereby made by completing the blank space in the 4th paragraph following the word "By" and before the words "the licensee may file" by inserting "November 15, 1975".

Dated at Bethesda, Maryland, this 17th day of October, 1975.

FOR THE NUCLEAR REGULATORY COMMISSION

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

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