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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:

Enclosed is a complete copy of the correspondence related to Amendment No. 14 which was distributed. Included are pages 4.5-12 and 4.5-5b which were inadvertently omitted in the Standby Gas Treatment System Technical Specification change that was received by Jersey Central Power and Light Company.

Sincerely,

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

Enclosure:
As stated

cc:
See next page

OFFICE ➤	ORB#3	ORB#3	ORB#3			
SURNAME ➤	CParrish CP	JShea:acr JJS	GLear GL			
DATE ➤	11/ 2 /76	11/ 2 /76	11/ 2 /76			

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

cc: G. F. Trowbridge, Esquire
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UNITED STATES
NUCLEAR REGULATORY COMMIS N
WASHINGTON, D. C. 20555

March 22, 1976

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:

In response to your request dated June 30, 1975, and revised by your letter dated December 17, 1975, the Commission has issued the enclosed Amendment No. 14 to Provisional Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station.

The amendment consists of changes in the Technical Specifications that modify the limiting conditions for operation and surveillance requirements for installed filters in the standby gas treatment system.

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

Sincerely,

George Lear

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

Enclosures:
1. Amendment No. 14 to License DPR-16
2. Safety Evaluation
3. Federal Register Notice

cc w/encls:
See next page

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. 14
License No. DPR-16

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

   A. The application for amendment by Jersey Central Power and Light Company (the licensee) dated June 30, 1975 and revised by your letter dated December 17, 1975, 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 activities will be conducted in compliance with the Commission's regulations; and

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

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

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment.

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

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

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

Attachment:
Changes to the
Technical Specifications

Date of Issuance: March 22, 1976

ATTACHMENT TO LICENSE AMENDMENT NO. 14

TO THE TECHNICAL SPECIFICATIONS

PROVISIONAL OPERATING LICENSE NO. DPR-16

DOCKET NO. 50-219

Replace page 1 of Table of Contents, and pages 1.0-5, 3.5-3, 3.5-4, 3.5-7, 4.5-5a, 4.5-6, 4.5-9a, and 4.5-9b, with the attached revised pages. Add pages 4.5-5b and 4.5-12. No change has been made on pages 3.4-4a, 3.5-5, 3.5-6, and 4.5-5.

TABLE OF CONTENTS

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

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.

1.23 STAGGERED TEST BASIS

A Staggered Test Basis shall consist of:

a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals.

b. The testing of one system, subsystem, train or other designated component at the beginning of each subinterval.

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. Two separate and independent standby gas treatment system circuits shall be operable when secondary containment is required except as specified by Specification 3.5.B.3.

3. With one standby gas treatment system circuit inoperable:

   a. During Power Operation:

      1. Demonstrate the operability of the other standby gas treatment system circuit within 2 hours, and

      2. Continue to demonstrate the operability of the standby gas treatment system circuit once per 24 hours until the inoperable standby gas treatment circuit is returned to operable status.

      3. Restore the inoperable standby gas treatment circuit to operable status within 7 days or be subcritical with reacto coolant temperature less than 212°F within the next 36 hour

   b. During Refueling:

      1. Demonstrate the operability of the redundant standby gas treatment system within 2 hours, and

      2. Continue to demonstrate the operability of the redundant standby gas treatment system once per 7 days until the inoperable system is returned to operable status.

      3. Restore the inoperable standby gas treatment system to operable status within 30 days or cease all spent fuel handling, core alterations or operations that could reduce the shutdown margin.

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.

es:

Specifications are placed on the operating states 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%Δk. 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).

The capacity of the fourteen suppression chamber to drywell vacuum relief valves is sized to limit the external pressure of the drywell during post-accident drywell cooling operations to the design limit of 2 psi. They are sized on the basis of the Bodega Bay pressure suppression tests.(9)(10) In Amendment 15 of the Oyster Creek FDSAR, Section II, the area of 2920 sq. in. is stated as the minimum area for flow of non-condensible gases from the suppression chamber to the drywell. To achieve this requirement, at least 12 of the 14 vacuum breaker valves (18" diameter) must be operable.

Each suppression chamber drywell vacuum breaker is fitted with a redundant pair of limit switches to provide fail safe signals to panel mounted indicators in the Reactor Building and alarms in the Control Room when the disks are open more than 0.1" at any point along the seal surface of the disk. These switches are capable of transmitting the disk closed-to-open signal with 0.01" movement of the switch plunger. Continued reactor operation with failed components is justified because of the redundancy of components and circuits and, most importantly, the accessibility of the valve lever arm and position reference external to the valve. The fail-safe feature of the alarm circuits assures operator attention if a line fault occurs.

Conservative estimates of the hydrogen produced, consistent with the core cooling system provided, show that the hydrogen air mixture resulting from a loss-of-coolant accident is considerably below the flammability limit and hence it cannot burn, and inerting would not be needed. However, inerting of the primary containment was included in the proposed design and operation. The 5% oxygen limit is the oxygen concentration limit stated by the American Gas Association for hydrogen-oxygen mixtures below which combustion will not occur

To preclude the possibility of starting up the reactor and operating a long period of time with a significant leak in the primary system, leak checks must be made when the system is at or near rated temperature and pressure. It has been shown(9)(10) that an acceptable margin with respect to flammability exists without containment inerting. Inerting the primary containment provides additional margin to that already considered acceptable. Therefore, permitting access to the drywell for the purpose of leak checking would not reduce the margin of safety below that considered adequate and is judged prudent in terms of the added plant safety offered by the opportunity for leak inspection. The 24-hour time to provide inerting is judged to be a reasonable time to perform the operation and establish the required O_2 limit.

Secondary containment[5] is designed to minimize any ground level release of radioactive materials which might result from a serious accident. The reactor building provides secondary containment during reactor operation when the drywell is sealed and in service and provides primary containment when the reactor is shut down and the drywell is open, as during refueling. Because the secondary containment is an integral part of the overall containment system, it is required at all times that primary containment is required. Moreover, secondary containment is required during fuel handling operations and whenever work is being performed on the reactor or its connected systems in the reactor building since their operation could result in inadvertent release of radioactive material.

The standby gas treatment system[6] filters and exhausts the reactor building atmosphere to the stack during secondary containment isolation conditions, with a minimum release of radioactive materials from the reactor building to the environs.

Two separate filter trains are provided each having 100% capacity.[6] If one filter train becomes inoperable, there is no immediate threat to secondary containment and reactor operation may continue while repairs are being made.

References

(1) FDSAR, Volume I, Section V-1
(2) FDSAR, Volume I, Section V-1.4.1
(3) FDSAR, Volume I, Section V-1.7
(4) Licensing Application, Amendment 11, Question III-25
(5) FDSAR, Volume I, Section V-2
(6) FDSAR, Volume I, Section V-2.4
(7) Licensing Application, Amendment 42
(8) Licensing Application, Amendment 32, Question 3
(9) Robbins, C. H., "Tests on a Full Scale 1/48 Segment of the Humboldt Bay Pressure Suppression Containment," GEAP-3596, November 17, 1960.
(10) Bodega Bay Preliminary Hazards Summary Report, Appendix 1, Docket 50-205, December 28, 1962.

4. Reactor Building to Suppression Chamber Vacuum Breakers

a. The reactor building to suppression chamber vacuum breakers and associated instrumention, including set point, shall be checked for proper operation every three months.

b. During each refueling outage each vacuum breaker shall be tested to determine that the force required to open the vacuum breaker from closed to fully open does not exceed the force specified in Specification 3.5.A.3.a. The air-operated vacuum breaker instrumentation shall be calibrated during each refueling outage.

5. Pressure Suppression Chamber - Drywell Vacuum Breakers

a. Periodic Operability Tests

Once each month and following any release of energy which would tend to increase pressure to the suppression chamber, each operable suppression chamber - drywell vacuum breaker shall be exercised. Operation of position switches, indicators and alarms shall be verified monthly by operation of each operable vacuum breaker.

b. Refueling Outage Tests

(1) All suppression chamber - drywell vacuum breakers shall be tested to determine the force required to open each valve from fully closed to fully open.

(2) The suppression chamber - drywell vacuum breaker position indication and alarm systems shall be calibrated and functionally tested.

(3) At least four of the suppression chamber - drywell vacuum breakers shall be inspected. If deficienci are found, all vacuum breakers shall be inspected and deficiencies corrected such that Specification 3.5.A.4.a can be met.

(4) A drywell to suppression chamber leak rate test shall demonstrate that with an initial differentia

pressure of not less th... 1.0 psi, the differential pressure decay rate shall not exceed the equivalent of air flow through a 2-inch orifice.

J. Reactor Building

1. Secondary containment capability tests shall be conducted after isolating the reactor building and placing either Standby Gas Treatment System filter train in operation.

2. The tests shall be performed at least once per operating cycle and shall demonstrate the capability to maintain a 1/4 inch of water vacuum under calm wind conditions with a Standby Gas Treatment System Filter train flow rate of not more than 4000 cfm.

3. A secondary containment capability test shall be conducted at each refueling outage prior to refueling.

4. The results of the secondary containment capability tests shall be in the subject of a summary technical report which can be included in the reports specified in Section 6.

K. Standby Gas Treatment System

1. The capability of each Standby Gas Treatment System circuit shall be demonstrated by:

a. At least once per 18 months, after every 720 hours of operation, and following significant painting, fire, or chemical release in the reactor building during operation of the Standby Gas Treatment System by verifying that:

(1) The charcoal adsorbers remove $\geq$99% of a halogenated hydrocarbon refrigerant test gas and the HEPA filters remove $\geq$99% of the DOP in a cold DOP test when tested in accordance with ANSI N510-1975.

(2) Results of laboratory carbon sample analysis show $\geq$90% radioactive methyl iodide removal efficiency when tested in accordance with ANSI N510-1975 (130°C, 95% relative humidity).

(3) Air distribution is uniform within $\pm$20% across the HEPA and charcoal filters.

b. At least once per 18 months by demonstrating:

(1) That the pressure drop across a HEPA filter is equal to or less than the maximum allowable pressure drop indicated in Figure 4.5.1.

(2) The inlet heater is capable of at least 10.9 KW input.

(3) Operation with a total flow within $\pm$ 10% of design flow.

c. At least once per 30 days on a STAGGERED TEST BASIS by operating each circuit for a minimum of 10 hours.

d. Anytime the HEPA filter bank or the charcoal adsorbers have been partially or completely replaced, the test for 4.5.K.1.a will be performed prior to returning the system to OPERABLE STATUS.

e. Automatic initiation of each circuit every 18 months.

L. Deleted

"M. Inerting Surveillance

When an inert atmosphere is required in the primary containment the oxygen concentration in the primary containment shall be checked at least weekly.

"N. Drywell Coating Surveillance

Carbon steel test panels coated with Firebar D shall be placed inside the drywell near the reactor core midplane level. They shall be removed for visual observation and weight loss measurements during the first, second, fourth and eighth refueling outages."

O. Instrument Line Flow Check Valves Surveillance

The capability of each instrument line flow check valve to isolate shall be tested at least once in every period between refueling outages. Each time an instrument line is returned to service after any condition which could have produced a pressure or flow disturbance in that line, the open position of the flow check valve in that line shall be verified. Such conditions include:

Leakage at instrument fittings and valves

Venting an instrument or instrument line

Isolating an instrument

Flushing or draining an instrument"

chamber pressure must not exceed a rate equivalent to the rate of air flow from the drywell to the suppression chamber through a 2-inch orifice. In the event the rate of change of pressure exceeds this value, then the source of leakage will be identified and eliminated before power operation is resumed.

The drywell-suppression chamber vacuum breakers are exercised monthly and immediately following termination of discharge of steam into the suppression chamber. This monitoring of valve operability is intended to assure that valve operability and position indication system performance does not degrade between refueling inspections. When a vacuum breaker valve is exercised through an opening-closing cycle, the position indicating lights are designed to function as follows:

Full Closed 2 Green - On
(Closed to $\leq$0.10" open) 2 Red - Off

Open 0.10" 2 Green - Off
(>0.10" open to full open) 2 Red - On

During each refueling outage, four suppression chamber-drywell vacuum breakers will be inspected to assure components have not deteriorated. Since valve internals are designed for a 40-year lifetime, an inspection program which cycles through all valves in about one-tenth of the design lifetime is extremely conservative. The alarm systems for the vacuum breakers will be calibrated during each refueling outage. This frequency is based on experience and engineering judgement.

Initiating reactor building isolation and operation of the standby gas treatment system to maintain a 1/4 inch of water vacuum, tests the operation of the reactor building isolation valves, leakage tightness of the reactor building and performance of the standby gas treatment system. Checking the initiating sensors and associated trip channels demonstrates the capability for automatic actuation. Performing these tests prior to refueling demonstrates secondary containment capability prior to extensive fuel handling operations associated with the outage. Testing prior to each refueling outage and between outages gives sufficient confidence of standby gas treatment system performance capability. A charcoal filter efficiency of 99% for halogen removal is adequate.

The in-place testing of charcoal filters is performed using Freon-112* which is injected into the system upstream of the charcoal filters. Measurement of the Freon concentration upstream and downstream of the charcoal filters is made using a gas chromatograph. The ratio of the inlet and outlet concentrations gives an overall indication of the leak tightness

Trade name of E. I. duPont de Nemours & Company of the system. Although this is basically a leak test, since the filters have charcoal of known efficiency and holding capacity for elemental iodine and/or methyl iodide, the test also gives an indicatation of the relative efficiency of the installed system. The test procedure is an adaptation of test procedures developed at the Savannah River Laboratory which were described in the Ninth AEC Air Cleaning Conference. The inlet heater input of 10.9 KW reduces the relative humidity of the air flow to ≤70% at 3000 cfm (i.e., assuming 100% relative humidity initially and 100°F).

High efficiency particulate filters are installed before and after the charcoal filters to minimize potential release of particulates to the environment and to prevent clogging of the iodine filters. An efficiency of 99% is adequate to retain particulates that may be released to the reactor building following an accident. This will be demonstrated by testing with DOP as testing medium.

If laboratory tests for the adsorber material in one circuit of the Standby Gas Treatment System are unacceptable, all adsorber material in that circuit shall be replaced with adsorbent qualified according to Regulatory Guide 1.52. Any HEPA filters found defective shall be replaced with those qualified with Regulatory Position C.3.d of Regulatory Guide 1.52.

*D. R. Muhbaier, "In Place Nondestructive Leak Test for Iodine Absorbers," Proceedings of the Ninth AEC Air Cleaning Conference, USAEC Report CONF-660904, 1966.

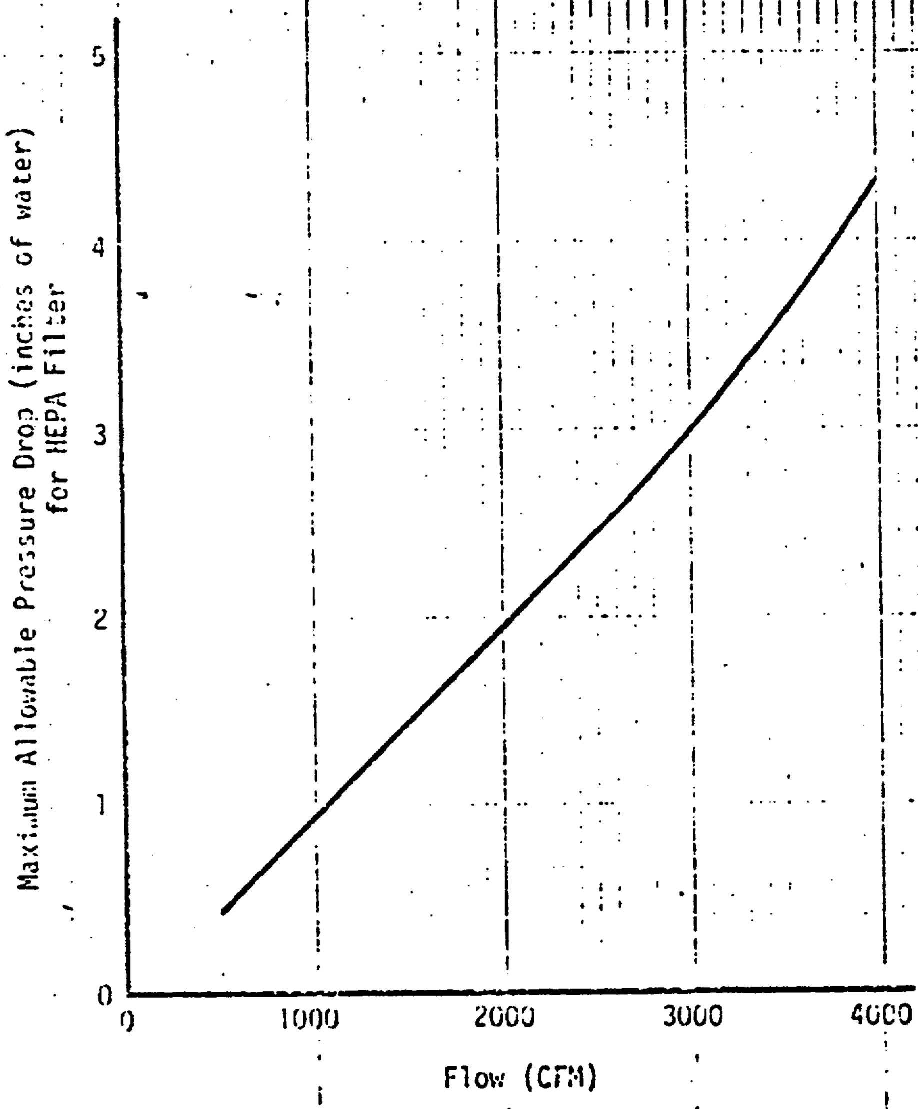
Maximum Allowable Pressure Drop (inches of water) for HEPA Filter
0
1
2
3
4
5
0
1000
2000
3000
4000
Flow (CFM)

FIGURE 4.5.1

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 14 TO PROVISIONAL OPERATING LICENSE NO. DPR-16

JERSEY CENTRAL POWER AND LIGHT COMPANY

DOCKET NO. 50-219

Introcution

By letter dated June 30, 1975, Jersey Central Power and Light Company (JCP&L) submitted an application for a license amendment which proposed to incorporate changes to the Oyster Creek Nuclear Generating Station Technical Specifications (Appendix A to the license). The amendment would modify limiting conditions for operation and surveillance requirements for installed filters in the standby gas treatment system. This submittal was in response to our request dated December 18, 1974. Based on discussions with us, JCP&L modified their original submittal by letter dated December 17, 1975.

Discussion

The standby gas treatment system (SGTS) provides a means for minimizing the release of radioactive material from the containment to the environs by first filtering and then exhausting the atmosphere from the reactor building. Primary containment atmosphere can also be directed to the SGTS for processing prior to release. Exhaust from the SGTS is discharged through the main off gas stack.

The SGTS consists of two identical, parallel air filtration trains. Each train has the capacity to clean up the reactor building atmosphere upon containment isolation and consists of a demister (moisture separator), prefilter, electrical heating coil, high efficiency particulate absorber (HEPA) filter, charcoal filter, HEPA filter, and exhaust fan.

The changes to the Technical Specifications proposed by JCP&L would provide additional limiting conditions for operation. The filters and fans of the SGTS are to undergo specific tests to assure their operating efficiency. The associated surveillance requirements provide the test intervals.

Evaluation

The Technical Specifications for installed filter systems as proposed by JCP&L are quite similar to those that we developed and transmitted to JCP&L in our letter of December 18, 1974. As stated in our letter, additional Limiting Conditions for Operation and Surveillance requirements are needed to ensure high confidence that the filter systems will function reliably, when needed, at a degree of efficiency equal to or better than that assumed in the accident analyses. Based on our review, we conclude that the changed Technical

Specification requirements will assure that the SGTS filter system will effectively reduce radioactive releases and will further assure that the releases during postulated accident conditions will not exceed the limits of 10 CFR Part 100.

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), than an environmental statement, negative declaration, or 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 the change 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 change 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.

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. 14 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 will modify limiting conditions for operation and surveillance requirements for installed filters in the standby gas treatment system.

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. Prior public notice of this amendment is 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 statement, negative declaration or environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the applications for amendment dated June 30, 1975, and December 17, 1975, (2) Amendment No. 14 to License No. DPR-16, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. and at the Ocean County Library, 15 Hooper Avenue, Toms River, New Jersey 08753.

A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 22 day of March 1976

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

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