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

LICENSE AUTHORITY FILE COPY

~~March 29, 1981~~

changed per 5/20/81 Correction Ltr. → May 8, 1981

Posted Amdt 54 to DPR-16

Docket No. 50-219
LS05-81-03-058

Mr. I.R. Finfrock, Jr.
Vice President
Jersey Central Power & Light Company
Oyster Creek Nuclear Generating Station
Post Office Box 388
Forked River, NJ 08731

Dear Mr. Finfrock:

SUBJECT: TMI-2 LESSONS LEARNED CATEGORY "A" ITEMS

The Commission has issued the enclosed Amendment No. 54 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 September 23, 1980.

The amendment approves changes to the Technical Specifications to incorporate certain TMI-2 Lessons Learned Category "A" requirements. These requirements concern (1) Emergency Power Supply/Inadequate Core Cooling, (2) Valve Position Indication, (3) Containment Isolation, (4) Shift Technical Advisor, (5) Integrity of Systems Outside Containment, and (6) Iodine Monitoring.

During our review of your application we found it necessary to modify four proposed Technical Specifications. We have discussed the changes with your representatives and have mutually agreed upon them.

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

Sincerely,

Dennis M. Crutchfield

Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
Division of Licensing

Enclosures:
See next page

Enclosures:
1. Amendment No. 54 to
   License No. DPR-16
2. Safety Evaluation
3. Notice of Issuance

cc w/enclosures:
See page 3

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

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 54
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 September 23, 1980, 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;

   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.

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 Appendices A and B, as revised through Amendment No. 54 , 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 M. Crutchfield

Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: ~~March 29, 1981~~
May 8, 1981
changed per 5/20/81 correction ltr.

TABLE OF CONTENTS

Amendment No. ~~14~~, ~~28~~, ~~29~~, 54

*Issued by NRC Order dated 10-24-80.

1.8 PLACE IN SHUTDOWN CONDITION

Proceed with and maintain an uninterrupted normal plant shutdown operation until the shutdown condition is met.

1.9 PLACE IN COLD SHUTDOWN CONDITION

Proceed with and maintain an uninterrupted normal plant shutdown operation until the cold shutdown condition is met.

1.10 PLACE IN ISOLATED CONDITION

Proceed with and maintain an uninterrupted normal isolation of the reactor from the turbine condenser system including closure of the main steam isolation valves.

1.11 REFUEL MODE

The reactor is in the refuel mode when the reactor mode switch is in the refuel mode position and there is fuel in the reactor vessel. In this mode the refueling platform interlocks are in operation.

1.12 REFUELING OUTAGE

For the purpose of designating frequency of testing and surveillance, a refueling outage shall mean a regularly scheduled refueling outage; however, where such outages occur within 8 months of the end of the previous refueling outage, the test or surveillance need not be performed until the next regularly scheduled outage. Following the first refueling outage, the time between successive tests or surveillance shall not exceed 20 months.

1.13 PRIMARY CONTAINMENT INTEGRITY

Primary containment integrity means that the drywell and absorption chamber are closed and all of the following conditions are satisfied:

A. All non-automatic primary containment isolation valves which are not required to be open for plant operation are closed.

B. At least one door in the airlock is closed and sealed.

C. All automatic containment isolation valves specified in Table 3.5.2 are operable or are secured in the closed position.

D. All blind flanges and manways are closed

1.14 SECONDARY CONTAINMENT INTEGRITY

Secondary containment integrity means that the reactor building is closed and the following conditions are met:

c. The reactor coolant system is maintained at less than 212 °F and vented.

d. At least one core spray pump, and system components necessary to deliver rated core spray flow to the reactor vessel, must remain operable to the extent that the pump and any necessary valves can be started or operated from the control room or from local control stations, and the torus is mechanically intact.

e. (1) No work shall be performed on the reactor or its connected systems which could result in lowering the reactor water level to less than 4'8" above the top of the active fuel and the condensate storage tank level is greater than thirty (30) feet (360,000 gallons). At least two redundant systems including core spray pumps and system components must remain operable as defined in d. above.

or

(2) The reactor vessel head, fuel pool gate, and separator-dryer pool gates are removed and the water level is above elevation 117 feet.

NOTE: When filling the reactor cavity from the condensate storage tank and draining the reactor cavity to the condensate storage tank, the 30 foot limit does not apply provided there is a sufficient amount of water to complete the flooding operation.

3. 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.

a) With one or more of the containment isolation valves shown in Table 3.5.2 inoperable:

1. Maintain at least one isolation valve operable in each affected penetration that is open and within 4 hours (48 hours for the traversing In-core probe system) either;

a) Restore the inoperable valve(s) to operable status or

b) Isolate each affected penetration by use of at least one deactivated automatic valve secured in the isolation position, or

c) Isolate each affected penetration by use of at least one closed manual valve or blind flange.

2. An inoperable containment isolation valve of the shutdown cooling system may be opened with a reator water temperature equal to or less than 350 °F in order to place the reactor in the

cold shutdown condition. The inoperable valve shall be returned to the operable status prior to placing the reactor in a condition where primary containment integrity is required.

TABLE 3.5.2

CONTAINMENT ISOLATION VALVES

VALVE FUNCTION/VALVE DESIGNATION	ISOLATION SIGNALS
Main Steam Isolation Valves (NS03A, NS03B, NS04A, NS04B)	1
Main Steam Condensate Drain Valves (V-1-106, V-1-107, V-1-110, V-1-111)	1
Reactor Building Closed Cooling Valves (V-5-147, V-5-166, V-5-167)	2
Instrument Air Valve (V-6-395)	1
Emergency Condenser Vent Valves (V-14-1, V-14-5, V-14-19, V-14-20)	1
Reactor Cleanup Valves (V-16-1, V-16-2, V-16-14, V-16-61)	3
Shutdown Cooling Valves (V-17-19, V-17-54)	3
Drywell Equipment Drain Tank Valves (V-22-1, V-22-2)	3
Drywell Sump Valves (V-22-28, V-22-29)	3
Drywell & Torus Atmosphere Control Valves (V-27-1, V-27-2, V-27-3, V-27-4, V-28-17, V-28-18, V-23-21, V-23-22, V-28-47, V-23-13, V-23-14, V-23-15, V-23-16, V-23-17, V-23-18, V-23-19, V-23-20)	3
Reactor Recirculation Loop Sample Valves (V-24-29, V-24-30)	1
Torus to Reactor Building Vacuum Relief Valves (V-26-16, V-26-18)	3*
Traversing In-Core Probe System (Tip machine ball valve No. 1, No. 2, No. 3, No. 4)	3

1)Reactor Isolation Signals as shown in Table 3.1.1
2)Low-Low Reactor Water Level and High Drywell Pressure; or Low-Low-Low Reactor Water Level.
3)Primary Containment Isolation Signals as shown in Table 3.1.1

*Valves automcatically reset to provide vacuum relief

(Correction)

3.13 ACCIDENT MONITORING INSTRUMENTATION

Applicability: Applies to the operating status of accident monitoring instrumentation.

Objective: To assure operability of accident monitoring instrumentation.

Specification: A. Relief and Safety Valve Position Indicators

1. The accident monitoring instrumentation channels show in Table 3.13.1 shall be operable when the mode switch is in the Startup or Run positions.

2. With the number of operable accident monitoring instrumentation channels less than the Total Number of Channels shown in Table 3.13.1, either restore the inoperable channels to operable status within 7 days, or place the reactor in the cold shutdown condition within 24 hours.

3. With the number of operable accident monitoring instrumentation channels less than the Minimum Channels Operable requirements of Table 3.13.1, either restore the inoperable channel(s) to the operable status within 48 hours, or place the reactor in the cold shutdown condition within 24 hours.

Bases:

The operability of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and access these variables during and following an accident. This capability is consistent with NUREG 0578.

TABLE 3.13.1

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
1. Relief and Safety Valve Position Indicator (Primary Detector*)	1/valve	1/valve
or Relief and Safety Valve Position Indicator (Backup indications**)	1/valve	

*Acoustic monitor

**Thermocouple

G. Continuous Leak Rate Monitor

1. When the primary containment is inerted the containment shall be continuously monitored for gross leakage by review of the inerting system makeup requirements.

2. This monitoring system may be taken out of service for the purpose of maintenance or testing but shall be returned to service as soon as practical.

H. Report of Test Results

Each integrated leakage rate test shall be the subject of a summary technical report, including results of the local leakage rate tests. The report shall include analysis and interpretation of the results which demonstrate compliance in meeting the specified leakage rate limits.

I. Functional Test of Valves

1. All containment isolation valves specified in Table 3.5.2 shall be tested for automatic closure by an isolation signal during each refueling otuage. The following valves are required to close in the time specified below:

Main steam line isolation valves ≥ 3 sec. and	≤10 sec.
Isolation condenser isolation valves	≤60 sec.
Cleanup system isolation valves	≤60 sec.
Cleanup auxiliary pumps system isolation valves	≤60 sec.
Shutdown system isolation valves	≤60 sec.

2. Each containment isolation valve shown in Table 3.5.2 shall be demonstrated operable prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator by cycling the valve through at least one complete cycle of full travel and verifying the specified isolating time. Following maintenance, repair or replacement work on the control or power circuit for the valves shown in Table 3.5.2, the affected component shall be tested to assure it will perform its intended function in the circuit.

3. During periods of sustained power operation each main steamline isolation valve shall be exercised in accordance with the following schedule.

a. Daily tests - Exercise valve (one at a time) to approximately 95% open position with reactor at operation power level.

b. Quarterly tests - Trip valve (one at a time) and check full closure time, with reactor power not greater than 50% of rated power.

4.13 ACCIDENT MONITORING INSTRUMENTATION

Applicability: Applies to surveillance requirements for the accident monitoring instrumentation.

Objective: To verify the operability of the accident monitoring instrumentation.

Specification:

A. Relief and Safety Valve Position Indicators

1. Each accident monitoring instrumentation channel shall be demonstrated operable by performance of the Channel Check and Channel Calibration operations at the frequencies shown in Table 4.13.1.

Bases:

The operability of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables during and following an accident. This capability is consistent with NUREG 0578.

TABLE 4.13-1

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Relief and Safety Valve Position Indicator (Primary Detector*)	A	B
Relief and Safety Valve Position Indicator (Backup Indications**)	A	B

Legend:

A = at least once per 31 days; B = at least once per 18 months (550 days).

*Acoustic Monitor

**Thermocouple

6.1 RESPONSIBILITY

6.1.1

The Director, Oyster Creek Operations shall be responsible for overall facility operation and shall delegate in writing the succession to this responsibility during his absence.

6.2 ORGANIZATION

OFFSITE

6.2.1

The offsite organization for technical support shall be as shown on Figure 6.2.1.

FACILITY STAFF

6.2.2

The facility organization shall be as shown on Figure 6.2.2 and:

a. Each on duty shift shall include at least the shift staffing indicated on Figure 6.2.2.

b. At least one licensed operator shall be in the control room when fuel is in the reactor.

c. Two licensed operators shall be in the control room during all reactor startups, shutdowns, and other periods involving planned control rod manipulations.

d. ALL CORE ALTERATIONS shall be directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.

e. An individual qualified in radiation protection measures shall be on site when fuel is in the reactor.

f. A Fire Brigade of at least 5 members shall be maintained onsite at all time. The Fire Brigade shall not include the minimum shift crew necessary for safe shutdown of the unit or any personnel required for other essential functions during a fire emergency.

g. Each on duty shift shall include a Shift Technical Advisor except that the Shift Technical Advisors position need not be filled if the reactor is in the refuel or shutdown mode and the reactor is less than 212°F.

plant. A maximum of four years of this five year experience may be fulfilled by related technical or academic training.

Shift Technical Advisor

Requirements: Bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant for transients and accidents.

6.3.2

Each member of the radiation protection organization for which there is a comparable position described in ANSI N18.1-1971 shall meet or exceed the minimum qualifications specified therein, or in the case of radiation protection technicians, they shall have at least one year's continuous experience in applied radiation protection work in a nucleaar facility dealing with radiological problems similar to those encountered in nuclear power stations, and shall have been certified by the Radiological Controls Manager, as qualified to perform assigned functions. This certification must be based on an NRC approved, documented program consisting of classroom training with appropriate examinations and documented positive findings by responsible supervision that the individual has demonstrated his ability to perform each specified procedure and assigned function with an understanding of its basis and purpose.

6.4 TRAINING

6.4.1

A retraining program for operators shall be maintained under the direction of the Manager, Training and shall meet the requirements and recommendation of Appendix A of 10CFR Part 55. Replacement training programs, the content of which shall meet the requirements of 10CFR Part 55, shall be conducted under the direction of the Manager, Training for licensed operators and Senior Reactor Operators.

6.4.2

A training program for the Fire Brigade shall be maintained under the direction of the Manager, Training.

6.5 SAFETY REVIEW AND AUDIT

The Director Oyster Creek Operations and three organizational units, the Plant Operations Review Committee (PORC), the Independent Safety Review Groups (ISRG) and the General Office Review Board (GORB) function to accomplish nuclear safety review and audit of the Oyster Creek Station.

6.5.1 Director Oyster Creek Operations (DOCO)

FUNCTION

6.5.1.1

The Director Oyster Creek Operations shall ensure that:

6.15 Integrity of Systems Outside Containment

The licensee shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. This program shall include the following:

1) Provisions establishing preventive maintenance and periodic visual inspection requirements, and

2) System leak test requirements, to the extent permitted by system design and radiological conditions, for each system at a frequency not to exceed refueling cycle intervals. The systems subject to this testing are (1) Core Spray, (2) Containment Spray, (3) Reactor Water Cleanup, (4) Isolation Condenser and (5) Shutdown Cooling.

6.16 Iodine Monitoring

The licensee shall implement a program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas* under accident conditions. This program shall include the following:

1. Training of personnel,
2. Procedures for monitoring, and
3. Provisions for maintenance of sampling and analysis equipment.

* Areas requiring personnel access for establishing hot shutdown condition.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

FOR OYSTER CREEK NUCLEAR GENERATING STATION

SUPPORTING AMENDMENT NO. 54 TO PROVISIONAL OPERATING LICENSE NO. DPR-16

JERSEY CENTRAL POWER & LIGHT COMPANY

DOCKET NO. 50-219

1.0 INTRODUCTION

By letter dated September 23, 1980, Jersey Central Power and Light Company (the licensee) submitted proposed changes to the Technical Specifications as contained in Appendix A to Provisional Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station. The proposed changes would incorporate certain TMI-2 Lessons Learned Category "A" requirements. The licensee's request is in response to the NRC staff's letter dated July 2, 1980.

2.0 DISCUSSION

By our letter dated September 13, 1979, we issued to all operating nuclear power plants requirements established as a result of our review of the TMI-2 accident. Certain of these requirements, designated Lessons Learned Category "A" requirements, were to have been completed by the licensee prior to any operation subsequent to January 1, 1980. Our evaluation of the licensee's compliance with these Category "A" items was enclosed with our letter to Jersey Central Power and Light Company dated May 8, 1980.

In order to provide reasonable assurance that operating reactor facilities are maintained within the limits determined acceptable following the implementation of the TMI-2 Lessons Learned Category "A" items, we requested that licensees amend their Technical Specifications to incorporate additional Limiting Conditions of Operation and Surveillance Requirements, as appropriate. This request was transmitted to all licensees on July 2, 1980. Included therein were model specifications that we had determined to be acceptable. The licensee's application is in response to our request. Each of the issues identified by the NRC staff and the licensee's response is discussed in the Evaluation below.

3.0 EVALUATION

3.1 Emergency Power Supply/Inadequate Core Cooling

As applicable to boiling water reactors, we indicated that water level instrumentation is important to post-accident monitoring and that surveillance of this instrumentation should be performed. The licensee's submittal did not specifically address this issue; however, the licensee indicated in discussions with the NRC staff that surveillance of the reactor water level instrumentation is included as part of the existing Technical Specification surveillance requirements for reactor water level protective instrumentation.

We reviewed Sections 3.1 and 4.1 of the existing Technical Specifications which address limiting conditions for operation and surveillance requirements for protective instrumentation and compared them against the guidelines for water level instrumentation in our model specifications. These guidelines, simply stated, require (1) two operable instrument channels; (2) with less than two channels operable, operability of two channels must be restored within seven days or reactor shutdown is required; (3) with less than one channel operable, operability must be restored within 48 hours; and (4) instrument checks should be performed once per month. The current technical specifications for the low and low-low reactor water level protective instrumentation requires that a minimum of two instrument channels be operable per operable trip system and that the control rods be inserted if the minimum conditions for operation are not met. In addition, channel checks are performed daily for this protective instrumentation.

Based on our review, we find that the current Oyster Creek Technical Specifications meet our guidelines. Therefore, we conclude that no further change is required.

3.2 Valve Position Indication

Our requirements for installation of a reliable position indicating system for relief and safety valves were based on the need to provide the operator with a diagnostic aid to reduce the ambiguity between indications that might indicate either an open relief/safety valve or a small line break. Such a system did not need to be safety grade provided that backup methods of determining valve position are available. The licensee's submittal included proposed technical specifications for relief valves but omitted specifications for safety valves. We discussed this matter with the licensee's representative who agreed that the safety valves should also be included in the proposed specifications. We also discussed clarification of "Primary Detector" and "Backup Indications" as indicated on proposed Tables 3.13.1 and 4.13.1. We mutually agreed that both tables should be interpreted so that "Primary Detector" means acoustic monitor, and "Backup Detector" means thermocouple.

Based on our review, we find that the licensee's submittal as modified by the NRC staff satisfies our requirements and is, therefore, acceptable.

3.3 Containment Isolation

Our request indicated that the Technical Specifications should include a Table of Containment Isolation Valves which reflect the diverse isolation signal requirement of this Lessons Learned issue. The licensee's response revises Section 3.5.A.3 of the Technical Specifications and provides a new Table 3.5.2 which lists the containment isolation valves. The licensee's response also revises Section 4.5.I by including references to the new Table 3.5.2.

We have reviewed the licensee's response which reflects diverse isolation signals to each valve. Based on our review, we find that the licensee's submittal satisfies our requirements and is acceptable.

3.4 Shift Technical Advisor

Our request indicated that the Technical Specifications related to shift manning should be revised to reflect the augmentation of a Shift Technical Advisor (STA). We discussed the current Technical Specifications with the licensee's representative and mutually agreed that the current specification 6.2.2.g should be clarified to indicate that each on duty shift shall include a Shift Technical Advisor except that this position need not be filled if the reactor is in the refuel or shutdown mode and the reactor water temperature is less than 212°F. The licensee's submittal indicates that the individual performing this function will have a Bachelor's Degree or equivalent in a scientific or engineering discipline with specific training in plant design, and responses and analysis of the plant for transients and accidents.

Based on our review, we find that the licensee's submittal as modified by the NRC staff satisfies our requirements and is, therefore, acceptable.

3.5 Integrity of Systems Outside Containment

Our request indicated that licensees should be required to periodically conduct a System Integrity Measurements Program to prevent the release of significant amounts of radioactivity to the environment via leakage from engineered safety systems and auxiliary systems which are located outside reactor containment. The licensee's response stated that they had reviewed the safety significance of the program and felt that it would be inappropriate to make it a license condition. The licensee did commit through responses to NUREG 0578 to initiate and maintain this program.

In discussions with the licensee's representative, we suggested and it was mutually agreed that it would be appropriate to include this requirement in the Administrative Controls Section of the Technical Specifications. The licensee's program includes provisions for a preventive maintenance program and periodic visual inspections. The program also includes system leak measurements at frequencies not to exceed refueling cycle intervals.

Based on our review, we find that the inclusion of this requirement in the Administrative Controls Section of the Technical Specifications satisfies our requirement and is, therefore, acceptable.

3.6 Iodine Monitoring

Our request indicated that the licensee should implement a program which will ensure the capability to determine the airborne iodine concentration in areas requiring personnel access under accident conditions. The licensee's response stated that they had reviewed the safety significance of the program and felt that it would be inappropriate to make it a license condition. The licensee did commit through responses to NUREG-0578 to intiate and maintain this program.

In discussions with the licensee's representative, we suggested and it was mutually agreed that it would be appropriate to include this requirement in the Administrative Controls Section of the Technical Specifications. The licensee's program includes training of personnel, procedures for monitoring, and provisions for maintenance of sampling and analysis equipment.

Based on our review we find that inclusion of this requirement in the Administrative Controls Section of the Technical Specifications satisfies our requirement and is, therefore, acceptable.

4.0 ENVIRONMENTAL CONSIDERATION

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.

5.0 CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) because the 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 rules and regulations and issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: March 29, 1981

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. 54 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. This amendment is effective as of its date of issuance.

The amendment approves Appendix A Technical Specifications for certain TMI-2 Lessons Learned Category "A" requirements concerning (1) Emergency Power Supply/Inadequate Core Cooling, (2) Valve Position Indication, (3) Containment Isolation, (4) Shift Technical Advisor, (5) Integrity of Systems Outside Containment, and (6) Iodine Monitoring.

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 action 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 September 23, 1980, (2) Amendment No. 54 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. 20555, 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) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 29th day of March, 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

Dennis M. Crutchfield

Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
Division of Licensing