

REGULATORY DOCKET FILE COPY

Docket No. 50-219

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

Dear Mr. Finfrock:

The Commission has issued the enclosed Amendment No. ⁴⁴ to Provisional Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station. This amendment consists of changes to the Technical Specifications and is in response to your application dated November 16, 1979.

This amendment modifies the definitions of shutdown and refueling condition in Section 1.0 of the Technical Specifications and eliminates unnecessary operability and surveillance requirements on specified protective instrumentation while in the shutdown, refuel and startup modes. The amendment also allows the pressure relief function of the electromagnetic relief valves to be inoperable or bypassed to permit the ASME Code System hydrostatic pressure tests at the end of the ten-year inspection interval. Typographical errors have been corrected and Technical Specifications Sections 2.1 and 3.5 have been clarified.

In reviewing your application certain corrections and clarifications, in addition to those proposed, were found to be necessary. We have discussed these changes with representatives of your staff and they are agreeable to the changes.

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

Sincerely,

Original signed by
 Dennis L. Ziemann

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

CPZ

Enclosures and cc:
 See next page

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DATE	1/2/80	1/2/80	1/2/80	1/4/80	1/3/80	1/3/80

Mr. I. R. Finrock, Jr.

- 2 -

January 4, 1980

Enclosures:

1. Amendment No. 44 to License No. DPR-16
2. Safety Evaluation
3. Notice of Issuance

cc w/enclosures:

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Newark, New Jersey 07102

**W/cy of incoming dtd. 11/16/79



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. 44
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 November 16, 1979, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;
 - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
 - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such 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.

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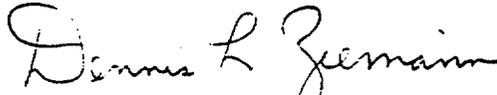
2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B. of Provisional Operating License No. DPR-16 is hereby amended to read as follows:

(B) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 44, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION



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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 4, 1980

ATTACHMENT TO LICENSE AMENDMENT NO. 44

PROVISIONAL OPERATING LICENSE NO. DPR-16

DOCKET NO. 50-219

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change.

PAGES

1.0-1

1.0-2

2.1-3

3.1-6*

3.1-6a

3.1-7

3.1-7a

3.1-8

3.1-9

3.1-10

3.1-11

3.1-12

3.1-12a

3.4-1b

3.5-2

3.7-1

3.7-2

*There are no changes to the provisions on this page. It has been retyped for administrative purposes only.

SECTION I

DEFINITIONS

The following frequently used terms are defined to aid in the uniform interpretation of the specifications.

1.1 OPERABLE

A system or component shall be considered operable when it is capable of performing its required function in its required manner.

1.2 OPERATING

Operating means that a system or component is performing its required function.

1.3 POWER OPERATION

Power operation is any operation when the reactor is in the startup mode or run mode except when primary containment integrity is not required.

1.4 STARTUP MODE

The reactor is in the startup mode when the reactor mode switch is in the startup mode position. In this mode, the reactor protection system scram trips initiated by condenser low vacuum and main steam line isolation valve closure are bypassed when reactor pressure is less than 600 psig; the low pressure main steamline isolation valve closure is bypassed; the IRM trips for rod block and scram are operable; and the SRM trips for rod block are operable.

1.5 RUN MODE

The reactor is in the run mode when the reactor mode switch is in the run mode position. In this mode, the reactor protection system is energized with APRM protection and the control rod withdrawal interlocks are in service.

1.6 SHUTDOWN CONDITION

The reactor is in a shutdown condition when the reactor mode switch is in the shutdown mode position and there is fuel in the reactor vessel. In this condition, the reactor is subcritical, a control rod block is initiated, all operable control rods are fully inserted, and specification 3.2-A is met.

1.7 COLD SHUTDOWN

The reactor is at cold shutdown when the mode switch is in the shutdown mode position, there is fuel in the reactor vessel, all operable control rods are fully inserted, and the reactor coolant system maintained at less than 212°F and vented.

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 adsorption 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 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:

The design basis critical heat flux correlation is based on an interrelationship of reactor coolant flow and steam quality. Steam quality is determined by reactor power, pressure, and coolant inlet enthalpy which in turn is a function of feedwater temperature and water level. This correlation is based upon experimental data taken over the entire pressure range of interest in a BWR, and the correlating line was determined by the statistical mean of the experimental data.

Curves are presented for two different pressures in Figure 2.1.1. The upper curve is based on nominal operating pressure of 1035 psia. The lower curve is based on a pressure of 1250 psia. In no case is reactor pressure ever expected to exceed 1250 psia because of protection system settings well below this value and, therefore, the curves will cover all operating conditions with interpolation. For pressures between 600 psia (the lower end of the critical heat flux correlation data) and 1035 psia, the upper curve is applicable with increased margin.

The power shape used in the calculation of Figure 2.1.1 is given in Table 3.2 of Reference 10 for a peak to average power of 1.5 with a peak location at the core midplane ($X/L = 0.5$). Table 3.2 further shows an axial power shape with an axial peak of the same magnitude but with a peak location above the core midplane ($X/L = 0.65$). These power shapes result in total peaking factors for each fuel type as shown in Specification 2.1.A.1. The total peaking factor for each fuel type is to be less than that specified in Section 2.1.A.1 at rated power. When operating below rated power with higher peaking factors as during control rod manipulation or near end of core life, applicability of the safety limit is assured by applying the reduction factors specified in 2.1.A.2.

The feedwater temperature assumed was the maximum design temperature output of the feedwater heaters at the given pressures and flows (e.g., 334°F at 1035 psia and 100% flow). For any lower feedwater temperature, subcooling is increased and the curves provide increased margin.

The water level assumed in the calculations was ten inches below the reactor low water level scram point (10'-7" above the top of the active fuel), which is the location of the bottom of the steam separator skirts. Of course, the reactor could not be operated in this condition. As long as the water level is above this point, the safety limit curves are applicable. As long as the water level is above the bottom of the steam separator skirts, the amount of carryunder would not be increased and the core inlet enthalpy would not be influenced.

The values of the parameters involved in Figure 2.1.1 can be determined from information available in the control room. Reactor pressure and flow are recorded and the APRM in-core nuclear instrumentation is calibrated in terms of percent power.

High flow in the main steamline is set at 120% of rated flow. At this setting the isolation valves close and in the event of a steam line break limit the loss of inventory so that fuel clad perforation does not occur. The 120% flow would correspond to the thermal power so this would either indicate a line break or too high a power.

Temperature sensors are provided in the steam line tunnel to provide for closure of the main steamline isolation valves should a break or leak occur in this area of the plant. The trip is set at 50°F above ambient temperature at rated power. This setting will cause isolation to occur for main steamline breaks which result in a flow of a few pounds per minute or greater. Isolation occurs soon enough to meet the criterion of no clad perforation.

The low-low-low water level trip point is set at 4'8" above the top of the active fuel and will prevent spurious operation of the automatic relief system. The trip point established will initiate the automatic depressurization system in time to provide adequate core cooling.

Specification 3.1.B.1 defines the minimum number of APRM channel inputs required to permit accurate average core power monitoring. Specifications 3.1.B.2 and 3.1.C.1 further define the distribution of the operable chambers to provide monitoring of local power changes that might be caused by a single rod withdrawal. Any nearby, operable LPRM chamber can provide the required input for average core monitoring. A Travelling Incore Probe or Probes can be used temporarily to provide APRM input(s) until LPRM replacement is possible. Since APRM rod block protection is not required below 61% of rated power, (1) as discussed in Section 2.3, Limiting Safety System Settings, operation may continue below 61% as long as Specification 3.1.B.1

and the requirements of Table 3.1.1 are met. In order to maintain reliability of core monitoring in that quadrant where an APRM is inoperable, it is permitted to remove the operable APRM from service for calibration and/or test provided that the same core protection is maintained by alternate means.

In the rare event that Travelling In-core Probes (TIPs) are used to meet the requirements 3.1.B or 3.1.C, the licensee may perform an analysis of substitute LPRM inputs to the APRM system using spare (non-APRM input) LPRM detectors and change the APRM system as permitted by 10 CFR 50.59.

Under assumed loss-of-coolant accident conditions it is inadvisable to allow the simultaneous starting of emergency core cooling and heavy load auxiliary systems in order to minimize the voltage drop across the emergency buses and to protect against a potential diesel generator overload. The diesel generator load sequence time delay relays provide this protective function and are set accordingly. The repetitive accuracy rating of the timer mechanism as well as parametric analyses to evaluate the maximum acceptable tolerances for the diesel loading sequence timers were considered in the establishment of the appropriate load sequencing.

Manual actuation can be accomplished by the operator and is considered appropriate only when the automatic load sequencing has been completed. This will prevent simultaneous starting of heavy load auxiliary systems and protect against the potential for diesel generator overload.

Reference:

(1)

NEDO-10189 "An Analysis of Functional Common Mode Failures in GE BWR Protection and Control Instrumentation", L. G. Frederick, et.al, July 1970.

Change No. ~~6~~
Amendment No. ~~9~~, ~~15~~, 44

TABLE 3.1.1 PROTECTIVE INSTRUMENTATION REQUIREMENTS

Function	Trip Setting	Reactor Modes in which Function Must Be Operable				Min. No. of Operable or Operating (Tripped) Trip Systems	Min. No. of Operable Instrument Channels Per Operable Trip Systems	Action Required*
		Shutdown	Refuel	Startup	Run			
A. Scram								
1. Manual Scram		X	X	X	X	2	1	Insert control rods
2. High Reactor Pressure	**		X(s)	X	X	2	2	
3. High Drywell Pressure	≤ 2 psig		X(u)	X(u)	X	2	2	
4. Low Reactor Water Level	**		X	X	X	2	2	
5. High Water Level in Scram Discharge Volume	≤ 37 gal.		X(a)	X	X	2	2	
6. Low Condenser Vacuum	≥ 23" Hg		X(b)	X(b)	X	2	2	
7. High Radiation in Main Steamline Tunnel	≤ 10 x normal background		X(s)	X	X	2	2	
8. Average Power Range Monitor (APRM)	**		X(c,s)	X(c)	X(c)	2	3	
9. Intermediate Range Monitor (IRM)	**		X(d)	X(d)		2	3	

TABLE 3.1.1 PROTECTIVE INSTRUMENTATION REQUIREMENTS

Function	Trip Setting	Reactor Modes in which Function Must be Operable				Min. No. of Operable or Operating (Tripped) Trip Systems	Min. No. of Operable Instrument Channels Per Operable Trip Systems	Action Required*
		Shutdown	Refuel	Startup	Run			
10. Main Steamline Isolation Valve Closure	**		X(b,s)	X(b)	X	2	4	Insert control rods
11. Turbine Trip Scram	**				X(j)	2	4	
12. Generator Load Rejection Scram	**				X(j)	2	2	

TABLE 3.1.1 PROTECTIVE INSTRUMENTATION REQUIREMENTS (CONTD)

Function	Trip Setting	Reactor Modes in Which Function Must Be Operable				Min. No. of Operable or Operating (Tripped) Trip Systems	Min. No. of Operable Instrument Channels Per Operable Trip Systems	Action Required*
		Shutdown	Refuel	Startup	Run			
B. <u>Reactor Isolation</u>								
1. Low-Low Reactor Water Level	**	X	X	X	X	2	2	Close main steam isolation valves and close isolation condenser vent valves, or place in cold shutdown condition
2. High Flow in Main Steamline A	\leq 120% rated	X (s)	X (s)	X	X	2	2	
3. High Flow in Main Steamline B	\leq 120% rated	X (s)	X (s)	X	X	2	2	
4. High Temperature in Main Steamline Tunnel	\leq Ambient at Power + 50°F	X (s)	X (s)	X	X	2	2	
5. Low Pressure in Main Steamline	**				X	2	2	
6. High Radiation in Main Steam Tunnel	\leq 10X Normal Background	X (s)	X (s)	X	X	2	2	
C. <u>Isolation Condenser</u>								
1. High Reactor Pressure	**	X (s)	X (s)	X	X	2	2	Place plant in cold shutdown condition
2. Low-Low Reactor Water	\geq 7' 2" above top of active fuel	X (s)	X (s)	X	X	2	2	

TABLE 3.1.1 PROTECTIVE INSTRUMENTATION REQUIREMENTS (CONTD)

Function	Trip Setting	Reactor Modes in Which Function Must be Operable				Min. No. of Operable or Operating (Tripped) Trip Systems	Min. No. of Operable Instrument Channels Per Operable Trip Systems	Action Required*
		Shutdown	Refuel	Startup	Run			
D. Core Spray								
1. Low-Low Reactor Water Level	**	X(t)	X(t)	X(t)	X	2	2	Consider the respective core spray loop in- operable, & com- ply with Spec. 3
2. High Drywell Pressure	≤ 2 psig	X(t)	X(t)	X(t)	X	2(k)	2(k)	
3. Low Reactor Pressure (valve permissive)	≥ 285 psig	X(t)	X(t)	X(t)	X	2	2	
E. Containment Spray								
1. High Drywell Pressure	≤ 2 psig	X(u)	X(u)	X(u)	X	2(k)	2(k)	Consider the con- tainment spray loop inoperable and comply with Spec. 3.4
2. Low-Low Reactor Water Level	> 7'2" above top of active fuel	X(u)	X(u)	X(u)	X	2	2	
F. Primary Containment Isolation								
1. High Drywell Pressure	≤ 2 psig	X(u)	X(u)	X(u)	X	2(k)	2(k)	Isolate contain- ment or place cold shutdown condition
2. Low-Low Reactor Water Level	> 7'2" above top of active fuel	X(u)	X(u)	X(u)	X	2	2	
G. Automatic Depressurization								
1. High Drywell Pressure	≤ 2 psig	X(v)	X(v)	X(v)	X	2(k)	2(k)	See note h

TABLE 3.1.1 PROTECTIVE INSTRUMENTATION REQUIREMENTS (CONTD)

Function	Trip Setting	Reactor Modes in Which Function Must Be Operable				Min. No. of Operable or Operating (Tripped) Trip Systems	Min. No. of Operable Instrument Channels Per Operable Trip Systems	Action Required*
		Shutdown	Refuel	Startup	Run			
2. Low-Low-Low Reactor Water Level	\geq 4'8" above top of active fuel	X(v)	X(v)	X(v)	X	2	2	See note h
3. AC Voltage	NA			X(v)	X	2	2	Prevent auto depressurization on loss of AC power. See note i
<u>II. Isolation Condenser Isolation</u>								Isolate Affected
1. High Flow Steam Line	\leq 20 psig Δ P	X(s)	X(s)	X	X	2	2	condenser, comply with Spec. 3.8
2. High Flow Con- densate Line	\leq 27" Δ P H ₂ O	X(s)	X(s)	X	X	2	2	
<u>I. Offgas System Isolation</u>								Isolate reactor
1. High Radiation in Offgas Line (e)	\leq 10 x Stack Release limit (See 3.6-A.1)	X(s)	X(s)	X	X	1	2	or trip the inoperable in- strument channel
<u>J. Reactor Building Isolation and Standby Gas Treatment System Initiation</u>								Isolate Reactor
1. High Radiation Reactor Building Operation Floor	\leq 100 Mr/Hr	X(w)	X(w)	X	X	1	1	Bldg. & Initiate Standby Gas Treat- ment System, or Manual Surveill- ance for not more
2. Reactor Bldg. Ventilation Exhaust	\leq 17 Mr/Hr	X(w)	X(w)	X	X	1	1	than 24 hours (total for all in- struments under
3. High Drywell Pressure	\leq 2 psig	X(u)	X(u)	X	X	1(k)	2(k)	J) in any 30-day period.
4. Low Low Reactor Water Level	\geq 7'2" above top of active fuel	X	X	X	X	1	2	

TABLE 3.1.1 PROTECTIVE INSTRUMENTATION REQUIREMENTS (CONTD)

Function	Trip Setting	Reactor Modes in Which Function Must Be Operable				Min. No. of Operable or Operating (Tripped) Trip Systems	Min.No.of Operable Instrument Channels Per Operable Trip Systems	Action Required*
		Shutdown	Refuel	Startup	Run			
<u>Rod Block</u>								
1. SRM Upscale	$\leq 5 \times 10^5$ cps		X	X(1)		1	3 (y)	No control rod withdrawals permitted
2. SRM Downscale	≥ 100 cps ^(f)		X	X(1)		1	3 (y)	
3. IRM Downscale	$\geq 5/125$ fullscale(g)		X	X		2	3	
4. APRM Upscale	**		X(s)	X	X	2	3(c)	
5. APRM Downscale	$\geq 2/150$ fullscale				X	2	3(c)	
6. IRM Upscale	$\leq 108/125$ fullscale		X	X		2	3	
<u>Condenser Vacuum Pump Isolation</u>								
1. High Radiation in Main Steam Tunnel	$\leq 10 \times$ Normal Background			During Startup and run when vacuum pump is operating		2	2	Insert control rods
<u>Diesel Generator Load Sequence Timers</u>								
1. Containment Spray Pump	Time delay after energiz. of relay 40 sec \pm 15%	X	X	X	X	2(m)	1(n)	Consider containment spray loop inoperable and comply with Spec. 3.4.C(See Note q)

TABLE 3.1.1 (CONT'D.)

- * Action required when minimum conditions for operation are not satisfied. Also permissible to trip inoperable trip system. When necessary to conduct tests and calibrations, one channel may be made inoperable for up to one hour per month without tripping its trip system.
- ** See Specification 2.3 for Limiting Safety System Settings.

Notes:

- a. Permissible to bypass, with control rod block, for reactor protection system reset in refuel mode.
- b. Permissible to bypass below 600 psig in refuel and startup modes.
- c. One (1) APRM in each operable trip system may be bypassed or inoperable provided the requirements of specification 3.1.C and 3.10.D are satisfied. Two APRM's in the same quadrant shall not be concurrently bypassed except as noted below or permitted by note .

Any one APRM may be removed from service for up to one hour for test or calibration without inserting trips in its trip system only if the remaining operable APRM's meet the requirements of specification 3.1.B.1 and no control rods are moved outward during the calibration or test. During this short period, the requirements of specifications 3.1.B.2, 3.1.C and 3.10.D need not be met.

- d. The (IRM) shall be inserted and operable until the APRM's are operable and reading at least 2/150 full scale.
- e. Air ejector isolation valve closure time delay shall not exceed 15 minutes.
- f. Unless SRM chambers are fully inserted.
- g. Not applicable when IRM on lowest range.
- h. One instrument channel in each trip system may be inoperable provided the circuit which it operates in the trip system is placed in a simulated tripped condition. If repairs cannot be completed within 72 hours the reactor shall be placed in the cold shutdown condition. If more than one instrument channel in any trip system becomes inoperable the reactor shall be placed in the cold shutdown condition. Relief valve controllers shall not be bypassed for more than 3 hours (total time for all controllers) in any 30-day period and only one relief valve controller may be bypassed at a time.

TABLE 3.1.1 (CON'D.)

- i. The interlock is not required during the start-up test program and demonstration of plant electrical output but shall be provided following these actions.
- j. Not required below 40% of turbine rated steam flow.
- k. All four (4) drywell pressure instrument channels may be made inoperable during the integrated primary containment leakage rate test (See Specification 4.5), provided that primary containment integrity is not required and that no work is 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.
- l. Bypassed in IRM Ranges 8, 9, & 10.
- m. There is one time delay relay associated with each of two pumps.
- n. One time delay relay per pump must be operable.
- o. There are two time delay relays associated with each of two pumps.
- p. Two time delay relays per pump must be operable.
- q. Manual initiation of affected component can be accomplished after the automatic load sequencing is completed.
- r. Time delay starts after closing of containment spray pump circuit breaker.
- s. These functions not required to be operable with the reactor temperature less than 212°F and the vessel head removed or vented.
- t. These functions may be inoperable or bypassed when corresponding portions in the same core spray system logic train are inoperable per Specification 3.4.A.
- u. These functions not required to be operable when primary containment integrity is not required to be maintained.
- v. These functions not required to be operable when the ADS is not required to be operable.
- w. These functions must be operable only when irradiated fuel is in the fuel pool or reactor vessel and secondary containment integrity is required per specification 3.5.B.
- y. The number of operable channels may be reduced to 2 per Specification 3.9-E and F.

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.

B. Automatic Depressurization System

1. Five electromatic relief valves of the automatic depressurization system shall be operable when the reactor water temperature is greater than 212°F and pressurized above 110 psig, except as specified in 3.4.B.2. The automatic pressure relief function of these valves (but not the automatic depressurization function) may be inoperable or bypassed during the system hydrostatic pressure test required by ASME Code Section XI, IS-500 at or near the end of each ten year inspection interval.

4. Reactor Building to Suppression Chamber Vacuum Breaker System

- a. Except as specified in Specification 3.5.A.4.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 of 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.

5. 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.

3.7 AUXILIARY ELECTRICAL POWER

Applicability: Applies to the operating status of the auxiliary electrical power supply.

Objective: To assure the operability of the auxiliary electrical power supply.

- Specification:
- A. The reactor shall not be made critical unless all of the following requirements are satisfied:
 1. The following buses or panels energized.
 - a. 4160 volt buses 1C and 1D in the turbine building switchgear room.
 - b. 460 volt buses 1A2, 1B2, 1A21, 1B21 vital MCC 1A2 and 1B2 in the reactor building switch gear room; 1A3 and 1B3 at the intake structure; 1A21A, 1B21A, 1A21B, and 1B21B and isolation valve MCC 1AB2 on 23'6" elevation in the reactor building; 1A24 and 1B24 at the stack.
 - c. 208/120 volt panels 3, 4, 4A, 4B, 4C and VACP-1 in the reactor building switchgear room.
 - d. 120 volt protection panel 1 and 2 in the cable room.
 - e. 125 volt DC distribution centers A and B, and panel D in the battery room; isolation valve motor control center DC-1 on 23'6" elevation in reactor building and panel E in the cable room.
 - f. 24 volt D.C. power panels A and B in the cable room.
 2. One 230 KV line is fully operational and switch gear and both startup transformers are energized to carry power to the station 4160 volt AC buses and carry power to or away from the plant.
 3. An additional source of power consisting of one of the following is in service connected to feed the appropriate plant 4160 V bus or buses:
 - a. A second 230 KV line fully operational.
 - b. One 34.5 KV line fully operational.
 4. The station batteries are available for normal service and a battery charger is in service for each battery, except one battery and associated charger may be removed from service as required for surveillance testing as set forth in Specification 4.7.B.
 - B. The reactor shall be placed in the cold shutdown condition if the availability of power falls below that required by Specification A above, except that the reactor may remain in operation for a period not to exceed 7 days in any 30 day period if a startup transformer is out of service.

None of the engineered safety feature-equipment fed by the remaining transformer may be out of service.

C. Standby Diesel Generators

1. The reactor shall not be made critical unless both diesel generators are operable and capable of feeding their designated 4160 volt buses.
2. If one diesel generator becomes inoperable during power operation, repairs shall be initiated immediately and the other diesel shall be operated at least one hour every 24 hours at greater than 20% rated power until repairs are completed. The reactor may remain in operation for a period not to exceed 7 days in any 30-day period if a diesel generator is out of service. During the repair period none of the engineered safety features normally fed by the operational diesel generator may be out of service or the reactor shall be placed in the cold shutdown condition.
3. If both diesel generators become inoperable during power operation, the reactor shall be placed in the cold shutdown condition.
4. For the diesel generators to be considered operable there shall be a minimum of 14,500 gallons of diesel fuel in the standby diesel generator fuel tank.

Bases:

The general objective is to assure an adequate supply of power with at least one active and one standby source of power available for operation of equipment required for a safe plant shutdown, to maintain the plant in a safe shutdown condition and to operate the required engineered safety feature equipment following an accident.

AC power for shutdown and operation of engineered safety feature equipment can be provided by any of four active (two 230 KV and two 34.5 KV lines) and either of two standby (two diesel generators) sources of power. Normally all six sources are available. However, to provide for maintenance and repair of equipment and still have redundancy of power sources the requirement of one active and one standby source of power was established. The plant's main generator is not given credit as a source since it is not available during shutdown. The plant 125V DC power is normally supplied by two batteries, each with an associated charger. A third charger is available to supply either battery. These chargers are active sources and supply the normal 125V DC requirements with the batteries as standby sources.

In applying the minimum requirement of one active and one standby source of AC power, since both 230 KV lines are on the same set of towers, either one or both 230 KV lines are considered as a single active source.



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 44 TO PROVISIONAL OPERATING LICENSE NO. DPR-16

JERSEY CENTRAL POWER & LIGHT COMPANY

OYSTER CREEK NUCLEAR GENERATING STATION

DOCKET NO. 50-219

1.0 Introduction

By letter dated November 16, 1979, Jersey Central Power and Light Company (JCP&L) (the licensee) requested changes to Appendix A, Technical Specifications, of Provisional Operating License No. DPR-16. Generally, these changes would remove unnecessary operability and surveillance requirements.

2.0 Evaluation

Definitions of Shutdown Condition, Cold Shutdown and Refuel Mode are being modified to reflect that they are applicable only when there is fuel in the reactor vessel. These definitions are used to define when various plant systems, equipment and instrumentation are required to be operable in order to assure that the plant is maintained in a safe condition. With the core unloaded, no equipment associated with reactor safety is required. The licensee is developing the administrative procedures to assure that the core is fully unloaded upon completion of defueling. These procedures will be available for review by Inspection and Enforcement prior to unloading. We find the proposed modifications to the definitions acceptable.

Protective instrumentation requirements are being revised to delete or otherwise modify the operability requirements of various instrumentation in the shutdown, refuel and startup modes. A detailed description of the changes is contained in the licensee's submittal. Our review of these changes reveal that these protective instrumentation functions do not provide any useful protection with the reactor in the specific operational modes. These functions generally provide protection when the reactor is pressurized and at power conditions. The exceptions to these protective functions are the SRM and building radiation instrumentation. Changes to all these instrumentation specifications are to clarify operability requirements and are consistent with the instrumentation functions. On this basis the proposed modifications are acceptable.

The licensee has also proposed to allow the pressure relief function of the electromatic relief valves to be inoperable or bypassed (the ADS function of the valves would be maintained) during the system hydrostatic pressure test required by AMSE Code Section XI, IS-500 at or near the end of each ten year inspection interval. This allowance is necessary since the hydrostatic test pressure is above the setpoint of the relief valves.

Even though the pressure relief function of the electromatic relief valves is bypassed, over pressure protection would continue to be provided by the 16 safety valves. Elimination of this relief function does not affect the reactor safety analyses, since credit was not taken for the relief function. Therefore, we find the modification acceptable.

In addition, the licensee has proposed to correct several typographical errors and to clarify several sections of the Technical Specifications. We have reviewed these changes and find them acceptable. Finally, we and the licensee have agreed to corrections and clarifications in addition to those proposed in the licensee's submittal. The most significant of these changes on page 3.1-12a is to clarify that core spray system operability requirements are for a complete single logic train, so that at least one train of the core spray system is available at required conditions. The first 1 1/2 paragraphs were repeated on page 3.1-6a and two words were misspelled on page 3.1-12a. These administrative errors were corrected. On this basis, we find the proposed modification acceptable.

3.0 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.

4.0 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 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: January 4, 1980

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

The amendment revises the definitions of shutdown and refuel condition in Section 1.0 of the Technical Specifications and eliminates unnecessary operability and surveillance requirements on specified protective instrumentation while in the shutdown, refuel, and startup modes. The amendment also allows the pressure relief function of the electromatic relief valves to be inoperable or bypassed to permit the ASME Code System hydrostatic pressure tests at the end of the ten-year inspection interval. Typographical errors have been corrected and Technical Specifications Sections 2.1 and 3.5 have been clarified.

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

- 2 -

For further details with respect to this action, see (1) the application for amendment dated November 16, 1979, (2) Amendment No. 44 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, 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 Operating Reactors.

Dated at Bethesda, Maryland, this 4th day of January, 1980.

FOR THE NUCLEAR REGULATORY COMMISSION



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

Regulatory Preparation for Jan. 1980 refueling - sections 1.0, 2.1, 3.1, 3.4, 5 and 3.7

William O. Miller, Chief
License Fee Management Branch, ADM

Date: 1/4/80

Amended Form Date:

(Preliminary and Final)

FACILITY AMENDMENT CLASSIFICATION - DOCKET NO(S). 50-219

Licensee: Jersey Central Power & Light Co.

Plant Name and Unit(s): Oyster Creek

License No(s): DPR-16

Mail Control No: 7911200533

Request Dated: 11-16-79

Fee Remitted: Yes No

Assigned TAC No:

Licensee's Fee Classification: Class I , II , III , IV , V , VI , None

Subject:

Amendment No. 44

Date of Issuance 1/4/80

- 1. This request has been reviewed by DCR/DPM in accordance with Section 170.22 of Part 170 and is properly categorized.
- 2. This request is incorrectly classified and should be properly categorized as Class _____. Justification for classification or reclassification: _____
- 3. Additional information is required to properly categorize the request: _____
- 4. This request is a Class _____ type of action and is exempt from fees because it:
 - (a) _____ was filed by a nonprofit educational institution,
 - (b) _____ was filed by a Government agency and is not for a power reactor,
 - (c) _____ is for a Class _____ (can only be a I, II, or III) amendment which results from a written Commission request dated _____ for the application and the amendment is to simplify or clarify license or technical specifications, has only minor safety significance, and is being issued for the convenience of the Commission, or
 - (d) _____ other (state reason therefor): _____

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Dennis L Zerrmann

Division of Operating Reactors/Project Management

- THE INITIAL FEE DETERMINATION HAS BEEN REASSESSED AND IS HEREBY AFFIRMED _____
- The above request has been reviewed and is exempt from fees.

LFMB 6/78

William O. Miller, Chief
License Fee Management Branch

Date

Attached: incoming and LA Amendment