JERSEY CENTRAL POWER & LIGHT COMPANY

OYSTER CREEK NUCLEAR GENERATING STATION

Provisional Operating
License No. DPR-16

Technical Specification
Change Request No. 75
Docket No. 50-219

Applicant submits, by this Technical Specification Change Request No. 75 to the Oyster Creek Nuclear Generating Station Technical Specifications, revised sections 1.0, 2.1, 3.1, 3.4, 3.5, and 3.7.

JERSEY CENTRAL POWER & LIGHT OCMPANY

BY: Ivan R. Finfrock
Vice President

STATE OF NEW JERSEY
COUNTY OF MORRIS

Sworn and subscribed to before me on this 16th day of November, 1979.

Phyllis A. Kabis
Notary Public

PHYLLIS A. KABIS
NOTARY PUBLIC OF NEW JERSEY
My Commission Expires Aug. 16, 1984

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Jersey Central Power & Light Company
Madison Avenue at Punch Bowl Road
Morristown, New Jersey 07960
(201) 455-8200

November 16, 1979

The Honorable Mary Lou Smith
Mayor of Lacey Township
P. O. Box 475
Forked River, New Jersey 08731

Dear Mayor Smith:

Enclosed herewith is one copy of Technical Specification Change Request No. 75 for the Oyster Creek Nuclear Generating Station Technical Specifications.

These documents were filed with the U. S. Nuclear Regulatory Commission on November 16, 1979.

Very truly yours,

Ivan R. Finfrock, Jr.
Vice President

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF)
) DOCKET NO. 50-219
JERSEY CENTRAL POWER & LIGHT COMPANY)

CERTIFICATE OF SERVICE

This is to certify that a copy of Technical Specification Change Request No. 75 for the Oyster Creek Nuclear Generating Station Technical Specifications, filed with the U. S. Nuclear Regulatory Commission on November 16 , 1979, has this 16th day of November, 1979, been served on the Mayor of Lacey Township, Ocean County, New Jersey by deposit in the United States mail addressed as follows:

The Honorable Mary Lou Smith
Mayor of Lacey Township
P. O. Box 475
Forked River, New Jersey 08731

JERSEY CENTRAL POWER & LIGHT COMPANY

BY: Ivan R. Finfrock
Vice President

DATED: November 16, 1979

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JERSEY CENTRAL POWER & LIGHT COMPANY
OYSTER CREEK NUCLEAR GENERATING STATION
PROVISIONAL OPERATING LICENSE NO. DPR-16
DOCKET NO. 50-219

Applicant hereby requests the Commission to change Appendix A to the License as follows:

1. Sections to be changed:

   1.0, 2.1, 3.1, 3.4, 3.5, 3.7.

2. Extent of Changes:

   The definitions of the shutdown and refueling conditions are modified in Section 1.0. Unnecessary operability requirements on protective instrumentation are deleted from Section 3.1. The operability requirements of the startup transformers are clarified in Section 3.7. Typographical errors in Sections 2.1 and 3.5 are corrected. The electromatic relief valves' pressure relief function is allowed to be inoperable during ASME Code System hydrostatic pressure tests.

3. Changes requested:

   The requested changes are on the attached revised Technical Specification pages 1.0-1, 1.0-2, 2.1-3, 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 and 3.7-2.

4. Discussion and Safety Evaluation:

   Definitions 1.6 Shutdown Condition, 1.7 Cold Shutdown and 1.11 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. When the core has been unloaded into the spent fuel pool, no equipment associated with reactor safety is required. A footnote has been added to Table 3.1.1 to require that Reactor Building isolation and Standby Gas Treatment System initiation on high radiation on the Reactor Building operating floor and on high radiation in the Reactor Building Ventilation Exhaust be operable when fuel is in the spent fuel pool or reactor vessel and secondary containment integrity is required per Specification 3.5.B.

   A change on page 2.1-3 is being made to correct a typographical error. The reference to "Reference 12" in the third paragraph should be to reference 10 (on page 2.1-5).

   Table 3.1.1 is being revised to delete or otherwise modify the operability requirements of various items of protective instrumentation in the shutdown, refuel and startup modes.

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The requirements to have the following protective instrumentation functions operable during the shutdown or refuel modes with the reactor water less than 212°F and the vessel head removed or vented have been deleted:

- A.2 High Reactor Pressure Scram
- A.7 High Radiation in Main Steamline Tunnel Scram
- A.8 Average Power Range Monitor Scram
- A.10 Main Steamline Isolation Valve Closure Scram
- B.2 High Flow in Main Steamline A Reactor Isolation
- B.3 High Flow in Main Steamline B Reactor Isolation
- B.4 High Temperature in Main Steamline Tunnel Reactor Isolation
- B.6 High Radiation in Main Steam Tunnel Reactor Isolation
- H.1 High Flow I. C. Steam Line - I. C. Isolation
- H.2 High Flow I. C. Condensate Line - I. C. Isolation
- I.1 High Radiation in Offgas Line - Offgas System Isolation
- K.4 APRM Upscale Rod Block

The third paragraph of Note C on Table 3.1.1 is being deleted. With the above change to A.8 the paragraph is no longer necessary.

With the reactor in shutdown or refuel, cooled down and vented, none of the above protective instrumentation functions provide any useful protection. The proposed change therefore in no way reduces the safety of the plant.

The requirements to have the following protective instrumentation functions operable are deleted during the indicated modes (Shutdown, Refuel, or Startup) when primary containment integrity is not required:

- A.3 High Drywell Pressure Scram (Refuel and Startup)
- E.1 High Drywell Pressure - Containment Spray (all 3 modes)
- E.2 Low-Low Reactor Water Level - Cont. Spray (all 3 modes)
- F.1 High Drywell Pressure - Primary Cont. Isolation (all 3 modes)
- F.2 Low-Low Water Level - Primary Cont. Isolation (all 3 modes)
- J.3 High Drywell Pressure - R. B. Isolation and SGTS Initiation (Shutdown and Refuel)

Specification 3.5.A.3 defines the need for primary containment integrity as follows:

> "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."

Since the reactor is not pressurized, no accident could occur which would result is high drywell pressure. Thus the high drywell pressure signals perform no useful function and need not be operable. For the same reason, containment spray is not required, nor should the instrumentation that initiates containment spray be required. When primary containment integrity is not being maintained, a primary containment isolation signal would serve no useful purpose and need not be operable since isolation valves may be inoperable, hatches may be open, etc. Isolation is provided instead by the secondary containment, which would be isolated by high radiation on the reactor building operating floor, high radiation in the reactor building ventilation exhaust, or low-low reactor water level.

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Table 3.1.1 items D.1, 2 and 3 are modified so that the protective instrumentation functions that initiate core spray may be inoperable or bypassed when the core spray system is inoperable as allowed by Specification 3.4.A. This does not affect the safety of the plant, since the instrumentation serves no useful purpose when the core spray system is inoperable.

Table 3.1.1 items G.1, 2 and 3 are modified so that these protective functions are not required when the Automatic Depressurization System is not required to be operable. Specification 3.4.B requires the ADS to be operable when the reactor water temperature is greater than 212°F and pressurized above 110 psig. When the ADS need not be operable, the functions that initiate ADS serve no useful purpose.

Table 3.1.1 items K.1 and 2 are modified to allow two operable SRM channels rather than three as permitted by Specifications 3.9.E and F. This change corrects an inconsistency that was overlooked when 3.9.E and F were issued in Amendment 23.

On page 3.5-2, Specification 3.5.A.4.a, the first line; the reference should be to 3.5.A.4.b rather than to 3.5.A.3.b. This was a typographical error.

The last sentence of Specification 3.7.B (page 3.7-1 and 2) is being modified to more accurately express its original intent. When one startup transformer is out of service, no engineered safety feature equipment associated with the remaining startup transformer may be out of service.

Section 3.4.B.1 is revised to allow the pressure relief function of the electromagnetic relief valves to 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. This allowance is necessary since the hydrostatic test pressure is above the setpoint of the relief valves. The ADS function of the valves would be maintained when the reactor water temperature is greater than 212°F and pressurized above 110 psig. Over-pressurization protection would continue to be provided by the safety valves.

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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.

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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:

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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.

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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 Instr-mentation", L. G. Frederick, et.al, July 1970.

TABLE 3.1.1 PROTECTIVE INSTRUMENTATION REQUIREMENTS

Function	Trip Setting	Reactor Modes in which Function Must Be Operable: Shutdown	Refuel	Startup	Run	Min. No. of Operable or Operating (Tripped) Trip Systems	Min. No. of Operable Instrument Channels Per Operable Trip Systems	Action Required*
A. Scram								Insert control rods
1. Manual Scram		X	X	X	X	2	1	
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	

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TABLE 3.1.1 PROTECTIVE INSTRUMENTATION REQUIREMENTS

Function	Trip Setting	Reactor Modes in which Function Must be Operable: Shutdown	Refuel	Startup	Run	Min. No. of Operable or Operating (Tripped) Trip Systems	Min. No. of Operable Instrument Channels Per Operable Trip Systems	Action Required*
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	

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TABLE 3.1.1 PROTECTIVE INSTRUMENTATION REQUIREMENTS (CONTD)

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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. Reactor Isolation								Close main steam isolation valves and close isolation condenser vent valves, or place in cold shutdown condition
1. Low-Low Reactor Water Level	**	X	X	X	X	2	2	
2. High Flow in Main Steamline A	≤ 120% rated	X (s)	X (s)	X	X	2	2	
3. High Flow in Main Steamline B	≤ 120% rated	X (s)	X (s)	X	X	2	2	
4. High Temperature in Main Steamline Tunnel	≤ 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	≤ 10X Normal Background	X (s)	X (s)	X	X	2	2	
C. Isolation Condenser								
1. High Reactor Pressure	**	X(s)	X(s)	X	X	2	2	Place plant in cold shutdown condition
2. Low-Low Reactor Water	≥ 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: Shutdown	Refuel	Startup	Run	Min. No. of Operable or Operating (Tripped) Trip Systems	Min. No. of Operable Instrument Channels Per Operable Trip Systems	Action Required*
D. Core Spray								Consider the respective core spray loop inoperable, & comply with Spec. 3.4
1. Low-Low Reactor Water Level	**	X(t)	X(t)	X(t)	X	2	2	
2. High Drywell Pressure	$\leq$ 2 psig	X(t)	X(t)	X(t)	X	2(k)	2(k)	
3. Low Reactor Pressure (valve permissive)	$\geq$ 285 psig	X(t)	X(t)	X(t)	X	2	2	
E. Containment Spray								
1. High Drywell Pressure	$\leq$ 2 psig	X(u)	X(u)	X(u)	X	2(k)	2(k)	Consider the containment spray loop inoperable and comply with Spec. 3.4
2. Low-Low Reactor Water Level	$\geq$ 7'2" above top of active fuel	X(u)	X(u)	X(u)	X	2	2	
F. Primary Containment Isolation								Isolate containment or place in cold shutdown condition
1. High Drywell Pressure	$\leq$ 2 psig	X(u)	X(u)	X(u)	X	2(k)	2(k)	
2. Low-Low Reactor Water Level	$\geq$ 7'2" above top of active fuel	X(u)	X(u)	X(u)	X	2	2	
G. Automatic Depressurization								See note h
1. High Drywell Pressure	$\leq$ 2 psig	X(v)	X (v)	X (v)	X	2(k)	2(k)	

TABLE 3.1.1 PROTECTIVE INSTRUMENTATION REQUIREMENTS (CONTD)

Function	Trip Setting	Reactor Modes in Which Function Must Be Operable: Shutdown	Refuel	Startup	Run	Min. No. of Operable or Operating (Tripped) Trip Systems	Min. No. of Operable Instrument Channels Per Operable Trip Systems	Action Required*
2. Low-Low-Low Reactor Water Level	≥ 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
H. Isolation Condenser Isolation								Isolate Affected isolation condenser, comply with Spec. 3.8
1. High Flow Steam Line	≤ 20 psig Δ P	X(s)	X(s)	X	X	2	2	
2. High Flow Condensate Line	≤ 27" Δ P H_2O	X(s)	X(s)	X	X	2	2	
I. Offgas System Isolation								
1. High Radiation in Offgas Line (e)	<10 x Stack Release limit (See 3.6-A.1)	X(s)	X(s)	X	X	1	2	Isolate reactor or trip the inoperable instrument channel
J. Reactor Building Isolation and Standby Gas Treatment System Initiation								Isolate Reactor Bldg. & Initiate Standby Gas Treatment System, or Manual Surveillance for not more han 24 hours (total for all instruments under J) in any 30-day period.
1. High Radiation Reactor Building Operation Floor	≤ 100 Mr/Hr	X(w)	X(w)	X	X	1	1	
2. Reactor Bldg. Ventilation Exhaust	≤ 17 Mr/Hr	X(w)	X(w)	X	X	1	1	
3. High Drywell Pressure	≤ 2 psig	X(u)	X(u)	X	X	1(k)	2(k)	
4. Low Low Reactor Water Level	≥ 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: Shutdown	Refuel	Startup	Run	Min. No. of Operable or Operating (Tripped) Trip Systems	Min. No. of Operable Instrument Channels Per Operable Trip Systems	Action Required*
Rod Block								No control rod withdrawals permitted
1. SRM Upscale	≤ 5 x 10^5 cps		X	X(1)		1	3 (y)	
2. SRM Downscale	≥ 100 cps(f)		X	X(1)		1	3 (y)	
3. IRM Downscale	≥ 5/125 fullscale(g)		X	X		2	3	
4. APRM Upscale	**		X (s)	X	X	2	3(c)	
5. APRM Downscale	≥ 2/150 fullscale				X	2	3(c)	
6. IRM Upscale	≤ 108/125 fullscale		X	X		2	3	
Condenser Vacuum Pump Isolation								Insert control rods
1. High Radiation in Main Steam Tunnel	≤ 10 x Normal Background		During Startup and run when vacuum pump is operating			2	2	
Diesel Generator Load Sequence Timers	Time delay after energiz. of relay							
1. Containment Spray Pump	40 sec ± 15%	X	X	X	X	2(m)	1(n)	Consider containment spray loop inoperable and comply with Spec. 3.4.C(See Note q)

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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 emperature less than 212°F and the vessel head removed or vented.

t. These functions may be inoperable or bypassed when corresponding portions of the core spray system are inoperable per Specification 3.4.A.

u. These functions not required to be operable when primary containment integrity is not requird 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.

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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.

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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.

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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 g ar 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.

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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.

JCP&L / GPU

Jersey Central Power & Light Company
Madison Avenue at Punch Bowl Road
Morristown, New Jersey 07960
(201) 455-8200

November 16, 1979

Director of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Sir:

Subject: Oyster Creek Generating Station
Docket No. 50-219
Technical Specification Change Request No. 75

In accordance with 10 CFR 50.59, Jersey Central Power & Light Company, owner and operator of the Oyster Creek Nuclear Generating Station, Provisional Operating License No. DPR-16, request changes to Appendix A of that license.

This Technical Specification Change will revise sections 1.0, 2.1, 3.1, 3.4, 3.5, and 3.7. These changes will facilitate the upcoming refueling outage (scheduled to start January 5, 1979) by removing technically unnecessary operability requirements and surveillance.

This Technical Specification Change Request has been reviewed and approved by the Station Manager, the Plant Operations Review Committee and an Independent Safety Review Group in accordance with Section 6.5 of the Oyster Creek Technical Specifications.

We have determined that this submittal is Class III per 10 CFR 170.22 and have enclosed a check for $4000 pursuant to that section.

Very truly yours,

Ivan R. Finfrock

Ivan R. Finfrock, Jr.
Vice President

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Enclosure

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JERSEY CENTRAL POWER & LIGHT COMPANY

OYSTER CREEK NUCLEAR GENERATING STATION

Provisional Operating
License No. DPR-16

Technical Specification
Change Request No. 75
Docket No. 50-219

Applicant submits, by this Technical Specification Change Request No. 75 to the Oyster Creek Nuclear Generating Station Technical Specifications, revised sections 1.0, 2.1, 3.1, 3.4, 3.5, and 3.7.

JERSEY CENTRAL POWER & LIGHT OCMPANY

BY: Ivan Finfrock
Vice President

STATE OF NEW JERSEY
COUNTY OF MORRIS

Sworn and subscribed to before me on this 16th day of November, 1979.

Phyllis A. Kabis
Notary Public

PHYLLIS A. KABIS
NOTARY PUBLIC OF NEW JERSEY
My Commission Expires Aug. 16, 1984

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