FEB 24 1976

Docket No. 50-219

DISTRIBUTION:
NRC PDR
Local PDR
Docket
ORB#3 Rdg
VStello
KRGoller/TJCarter
CParrish
WPaulson
OELD
OI&E (7)
BJones (4)
BScharf (15)
JMcGough
JSaltzman
ACRS (16)
OPA
TBAbernathy
JRBuchanan
Gray file
extra cps

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

Gentlemen:

In response to your request dated December 23, 1975 (as supplemented by your letter of February 5, 1976) and January 27, 1976, the Commission has issued the enclosed Amendment No. 15 to Provisional Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station.

The amendment consists of changes that will (1) revise the Technical Specifications relating to operating limits (maximum average planar linear heat generation rate) and operability requirements of the automatic depressurization system based on a reevaluation of the Oyster Creek Nuclear Generating Station ECCS performance, and (2) add limiting conditions for operation and surveillance requirements for the time delay relays which control the loading of the diesel generators under postulated loss-of-coolant accident conditions.

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

Sincerely,

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

Enclosures:
1. Amendment No. 15 to License DPR-16
2. Safety Evaluation
3. Federal Register Notice

cc: See next page

OFFICE	ORB#3	ORB#3	OELD	ORB#3	RL:AD/ORs	
SURNAME	CParrish:kmf	WPaulson		GLear	KRGoller	
DATE	2/20/76	2/20/76	2/23/76	2/23/76	2/24/76	

Jersey Central Power & Light Co. - - -

cc:

G. F. Trowbridge, Esquire
Shaw, Pittman, Potts and Trowbridge
Barr Building
910 17th Street, N. W.
Washington, D. C. 20006

Jersey Central Power & Light Company
ATTN: Mr. Thomas M. Crimmins, Jr.
Safety and Licensing Manager
GPU Service Corporation
260 Cherry Hill Road
Parsippany, New Jersey 07054

Anthony Z. Roisman, Esquire
Roisman, Kessler and Cashdan
1712 N Street, N. W.
Washington, D. C. 20036

Honorable Joseph W. Ferraro, Jr.
Deputy Attorney General
State of New Jersey
101 Commerce Street - Room 208
Newark, New Jersey 07102

George F. Kugler, Jr.
Attorney General
State of New Jersey
State House Annex
Trenton, New Jersey 08625

Ocean County Library
15 Hooper Avenue
Toms River, New Jersey 08753

The Honorable W. M. Mason
Mayor, Lacey Township
P. O. Box 475
Forked River, New Jersey 08731

Honorable Wm. F. Hyland
Attorney General
State of New Jersey
State House Annex
Trenton, New Jersey 08601

Steven P. Russo, Esquire
248 Washington Street
P. O. Box 1060
Toms River, New Jersey 08753

UNITED STATES
NUCL__R 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. 15
License No. DPR-16

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

   A. The applications for amendment by Jersey Central Power and Light Company (the licensee) dated December 23, 1975 (and supplement dated February 5, 1976) and January 27, 1976, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations; and

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

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

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and Paragraphs 3.E and 3.F of the license are hereby deleted in their entirety.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Karl R. Goller

Karl R. Goller, Assistant Director
for Operating Reactors
Division of Operating Reactors

Attachment:
Changes to the
Technical Specifications

Date of Issuance: February 24, 1976

ATTACHMENT TO LICENSE AMENDMENT NO. 15

TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-16

DOCKET NO. 50-219

Replace pages 3.1-6a, 3.1-11, 3.1-12a, 3.3-2, 3.4-2, 3.4-5, 4.7-1, 4.7-2 and Figure 3.10.1 with the attached revised pages. Add pages 3.1-11a, 3.3-2a and 3.10-5.

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.

Whenever it is necessary to replace an LPRM assembly, the operation requires the removal of fuel bundles in order to eliminate interference with the LPRM assembly. During the operation, the reactor mode switch will be locked in the REFUEL position in accordance with Technical Specification 3.9.B. In addition, the initial fuel loading non-coincidence jumpers in the Reactor Protective System will be removed. This provides additional protection for the core because any one out of four Source Range Monitor (SRM) channels or any one of eight APRM channels can produce a full scram (i.e., trip both Protection System Channels) if the flux reaches their respective setpoints.

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.

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*
K.	Rod Block								No control rod withdrawals permitted
	1. SRM Upscale	$\leq 5 \times 10^5$ cps		X	X(1)		1	3	
	2. SRM Downscale	$\geq$ 100 cps(f)		X	X(1)		1	3	
	3. IRM Downscale	$\geq$ 5/125 fullscale(g)		X	X		2	3	
	4. APRM Upscale	**		X	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	
L.	Condenser Vacuum Pump Isolation								Insert control rods
	1. High Radiation in Main Steam Tunnel	$\leq$ 10 x Normal Background		During Startup and run when vacuum pump is operating			2	2	
M.	Diesel Generator Load Sequence Timers	Time delay after energiz. of relay							
	1. Containment Spray Pump	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 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			
M. Diesel Generator Load Sequence Timers (Cont'd.)								
2. CRD pump	60 sec ± 15%	X	X	X	X	2(m)	1(n)	Consider the pump inoperable and comply with Spec. 3.4.D (See Note q).
3. Emerg. Service Water Pump (r)	45 sec ± 15%	X	X	X	X	2(m)	1(n)	Consider the loop inoperable and comply with Spec. 3.4.C (See Note q).
4. Service Water Pump	120 sec ± 15%	X	X	X	X	2(o)	2(p)	Consider the pump inoperable and comply within 7 days (See Note q).
5. Closed Cooling Water Pump	166 sec ± 15%	X	X	X	X	2(m)	1(n)	Consider the pump inoperable and comply within 7 days (See Note q).

TABLE 3.1.1 (CONTD.)

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 the plant is in the cold shutdown condition 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.

2. The reactor coolant quality shall not exceed the following limits during power operation with steaming rates to the turbine-condenser of at least 100,000 pounds per hour.

conductivity	10 μmho/cm
chloride ion	1.0 ppm

3. If Specification 3.3.E.1 and 3.3.E.2 cannot be met, the reactor shall be placed in the cold shutdown condition.

F. Recirculation Loop Operability

1. The reactor shall not be operated with one or more recirculation loops out of service except as specified in Specification 3.3.F.2.

2. Reactor operation with one idle recirculation loop is permitted provided that the idle loop is not isolated from the reactor vessel.

3. If Specifications 3.3.F.1 and 3.3.F.2 are not met the reactor shall be placed in the cold shutdown condition within 24 hours.

Bases:

The reactor coolant system [(1)] is a primary barrier against the release of fission products to the environs. In order to provide assurance that this barrier is maintained at a high degree of integrity, restrictions have been placed on the operating conditions to which it can be subjected.

The nil-ductility transition (NDT) temperature is defined as the temperature below which ferritic steel breaks in a brittle rather than ductile manner. Radiation exposure from fast neutrons (> 1 mev) above about 10^{17} nvt may increase the NDT temperature of the vessel base metal. Extensive tests have established the magnitude of changes in the NDT temperature as a function of the integrated neutron exposure [(2)]. Reference (2) presents pertinent test data for the type material (SA302B) [(3)] used as the base metal for this vessel. The upper two curves of Reference (2) apply to thick-walled pressure vessels and the lower curve is for the wall thickness range representative of this reactor vessel.

The initial NDT temperature of the vessel shell material opposite the reactor core region is 10°F and elsewhere is 40°F [(3,4)]. The design life of the reactor vessel is 40 years and the maximum fast neutron exposure at 40 years is calculated to be 1×10^{18} nvt. [(3)]

The NDT temperature limit curve in Figure 3.3.1 uses the "worst case" curve of Reference (2) to establish the NDT temperature shift and is, therefore, based on the more conservative thick-walled pressure vessel data. For example, the expected NDT temperature shift for this vessel at 10^{18} nvt is expected to be 15°F instead of the 90°F[2,3] assumed in establishing Figure 3.3.1. Figure 3.3.1 also incorporates a 60° factor of safety. This factor is based upon the requirements of the ASME code and the considerations[5] which resulted in these requirements. Therefore, the specification provides for "worst case" data as well as 60°F of margin to provide assurance that non-ductile failure will not occur.

2. If at any time there are only four operable electromatic relief valves, the reactor may remain in operation for a period not to exceed 3 days provided the motor operated isolation and condensate makeup valves in both isolation condensers are demonstrated daily to be operable.

3. If Specifications 3.4.B.1 and 3.4.B.2 are not met, reactor pressure shall be reduced to 110 psig or less, within 24 hours.

4. The time delay set point for initiation after coincidence of low-low-low reactor water level and high drywell pressure shall be set not to exceed two minutes.

C. Containment Spray System and Emergency Service Water System

1. The containment spray system and the emergency service water system shall be operable at all times with irradiated fuel in the reactor vessel, except as specified in Specifications 3.4.C.3, 3.4.C.4, 3.4.C.6 and 3.4.C.8.

2. The absorption chamber water volume shall not be less than 82,000 ft^3 in order for the containment spray and emergency service water systems to be considered operable.

3. If one emergency service water system loop becomes inoperable, its associated containment spray system loop shall be considered inoperable. If one containment spray system loop and/or its associated emergency service water system loop becomes inoperable during the run mode, the reactor may remain in operation for a period not to exceed 7 days provided the remaining containment spray system loop and its associated emergency service water system loop each have no inoperable components and are demonstrated daily to be operable.

4. If a pump in the containment spray system or emergency service water system becomes inoperable, the reactor may remain in operation for a period not to exceed 15 days provided the other similar pump is demonstrated daily to be operable. A maximum of two pumps may be inoperable provided the two pumps are not in the same loop. If more than two pumps become inoperable the limits of Specification 3.4.C.3 shall apply.

5. During the period when one diesel is inoperable, the containment spray loop and emergency service water system loop connected to the operable diesel shall have no inoperable components.

6. If primary containment integrity is not required (see Specification 3.5.A.1), the containment spray system may be made inoperable.

spray pump capable of full rated flow and the 72 hour operability demonstration of both core spray pumps is specified.

The relief valves of the automatic depressurization system enable the core spray system to provide protection against the small break in the event the feedwater system is not active.

The containment spray system is provided to remove heat energy from the containment in the event of a loss-of-coolant accident. The flow from one pump in either loop is more than ample to provide the required heat removal capability(2). The emergency service water system provides cooling to the containment spray heat exchangers and, therefore, is required to provide the ultimate heat sink for the energy release in the event of a loss-of-coolant accident. The emergency service water pumping requirements are those which correspond to containment cooling heat exchanger performance implicit in the containment cooling description. Since the loss-of-coolant accident while in the cold shutdown condition would not require containment spray, the system may be deactivated to permit integrated leak rate testing of the primary containment while the reactor is in the cold shutdown condition.

The control rod drive hydraulic system can provide high pressure coolant injection capability. For break sizes up to 0.002 ft^2, a single control rod drive pump with flow of 110 gpm is adequate for maintaining the water level nearly five feet above the core, thus alleviating the necessity for auto-relief actuation (3).

The core spray main pump compartments and containment spray pump compartments were provided with water-tight doors. (4) Specification 3.4.E ensures that the doors are in place to perform their intended function.
"Similarly, since a loss-of-coolant accident when primary containment integrity is not being maintained would not result in pressure build up in the drywell or torus, the system may be made inoperable under these conditions. This prevents possible personnel injury associated with contact with chromated torus water."

References

(1) Licensing Application, Amendment
(2) Licensing Application, Amendment 32, Question 3
(3) Licensing Application, Amendment 18, Question 1
(4) Licensing Application, Amendment 18, Question 4

Where: VF = Bundle average void fraction
PR = Assembly radial power factor
FCP = Fractional core power (relative to 1930 MWt)
B = Power-Void Limit

The limiting values of "B" for each fuel type are shown in the table below.

Fuel Type(s)	B
I, II, III	.365
IIIE, IIIF	.377
V, VB	.332

D. During steady state power operation, MCPR shall be $\geq$ 1.73 for 8x8 fuel and $\geq$ 1.69 for 7x7 fuel. If at any time during steady state power operation it is determined by normal surveillance that the limiting value for MCPR is being exceeded action shall then be initiated to restore operation to within the prescribed limits. Surveillance and corresponding action shall continue until the prescribed limits are again being met.

Basis: The specification for average planar LHGR assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the 2200°F limit specified in 10 CFR 50.46 (January 4, 1974) considering the postulated effects of fuel pellet densification.

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is only dependent secondarily on the rod to rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than ± 20°F relative to the peak temperature for a typical fuel design, the limit on the average linear heat generation rate is sufficient to assure that calculated temperatures are below the limits specified in 10 CFR 50.46 (January 4, 1974).

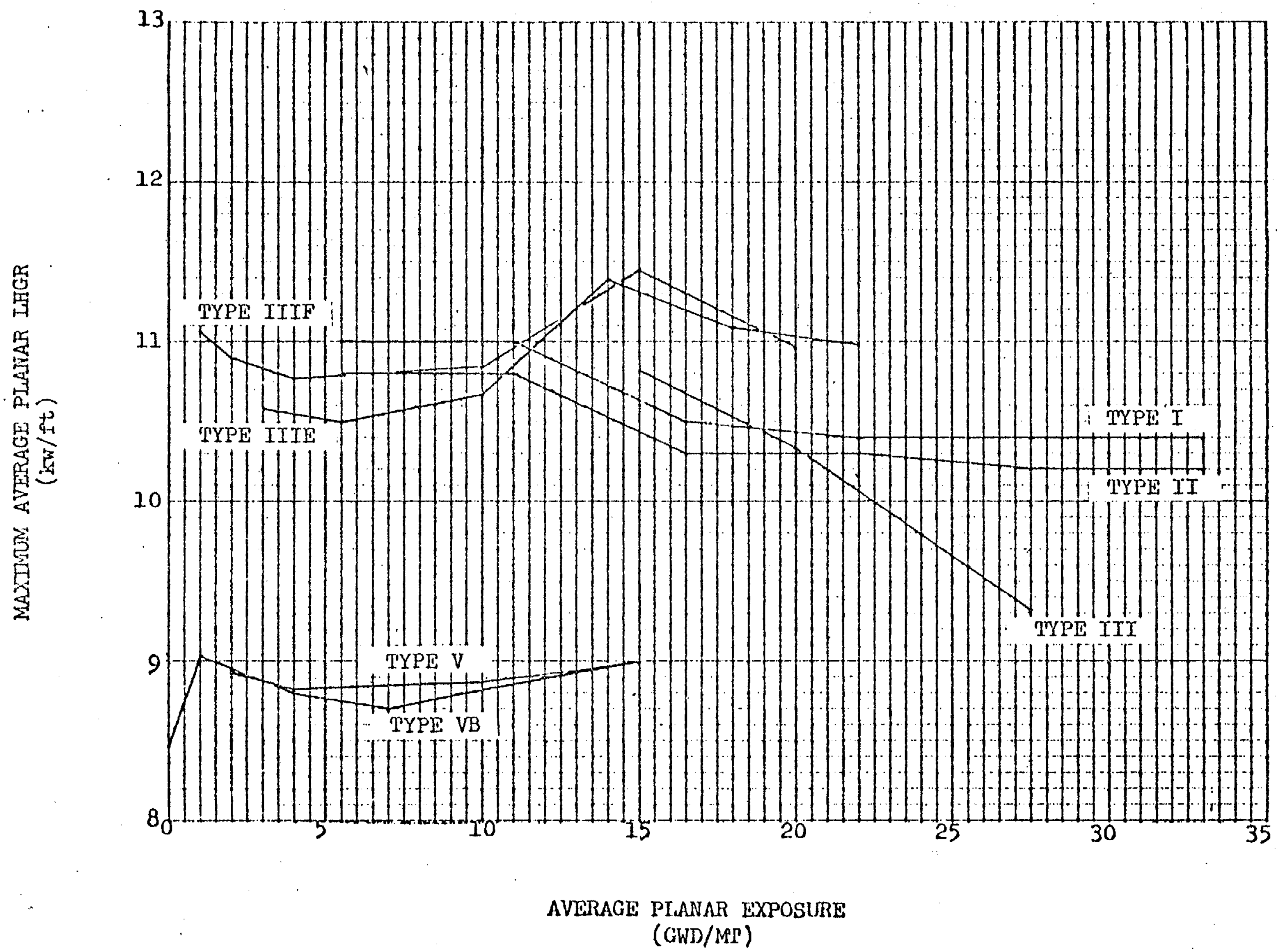
The maximum average planar LHGR shown in Figure 3.10.1 for Type I and II fuel are the result of LOCA analyses performed utilizing a blowdown thermal-hydraulic analysis developed by General Electric Company in compliance with 10 CFR 50, Appendix K (January 4, 1974). Single failure considerations were based on the revised Oyster Creek Single Failure Analysis submitted to the Staff on July 15, 1975.

The maximum average planar LHGR shown in Figure 3.10.1 for Type III, IIIE, IIIE, V and VB fuel are the result of Appendix K approved LOCA analyses performed by Exxon Nuclear Company utilizing blowdown results obtained from General Electric Company which reflect revised single failure considerations.(1)

The possible effects of fuel pellet densification are: (1) creep collapse of the cladding due to axial gap formation; (2) increase in the LHGR because of pellet column shortening; (3) power spikes due to axial gap formation; and (4) changes in stored energy due to increased radial gap size.

FIGURE 3.10.1

MAXIMUM ALLOWABLE AVERAGE PLANAR
LINEAR HEAT GENERATION RATE



REFERENCES

1. Oyster Creek Nuclear Generating Station, Loss-of-Coolant Accident Analysis Reevaluation and Technical Specification Change Request No. 42, Attachment I, dated December 23, 1975.

4.7 AUXILIARY ELECTRICAL POWER

Applicability: Applies to surveillance requirements of the auxiliary electrical supply.

Objective: To verify the availability of the auxiliary electrical supply.

Specification: A. Diesel Generator

1. Each diesel generator shall be started and loaded to not less than 20% rated power every two weeks.

2. The two diesel generators shall be automatically actuated and functionally tested during each refueling outage. This shall include testing of the diesel generator load sequence timers listed in Table 3.1.1.

3. Each diesel generator shall be given a thorough inspection at least annually.

4. The diesel generators' fuel supply shall be checked following the above tests.

5. The diesel generators' starting batteries shall be tested and monitored the same as the station batteries, Specification 4.7.b.

B. Station Batteries

1. The specific gravity and voltage of the designated pilot cell, the temperature of the adjacent cell, and the overall battery voltage shall be measured weekly.

2. The voltage of each cell shall be measured monthly to the nearest 0.01 volt.

3. The specific gravity of each cell, the temperature reading of every fifth cell, the height of electrolyte, and the amount of water added shall be measured every 3 months.

4. The batteries shall be load tested every 6 months, including monitoring the battery voltage as a function of time.

Basis: The biweekly tests of the diesel generators are primarily to check for failures and deterioration in the system since last use. The manufacturer has recommended the two week test interval, based on experience with many of their engines. One factor in determining this test interval (besides checking whether or not the engine starts and runs) is that the lubricating oil should be circulated through the engine approximately every two weeks. The diesels should be loaded to at least 20% of rated power until engine and generator temperatures have stabilized (about one hour). The minimum 20% load will prevent soot formation in the cylinders and injection nozzles. Operation up to an equilibrium temperature ensures that there is no over-heat problem. The tests also provide an engine and generator operating history to be compared with subsequent engine-generator test data to identify and correct any mechanical or electrical deficiency before it can result in a system failure.

The test during refueling outages is more comprehensive, including procedures that are most effectively conducted at that time. These include automatic actuation and functional capability tests, to verify that the generators can start and assume load in less than 20 seconds and testing of the diesel generator load sequence timers which provide protection from a possible diesel generator overload during LOCA conditions. The annual, thorough inspection will detect any signs of wear long before failure.

The manufacturer's instructions for battery care and maintenance with regard to the floating charge, the equalizing charge, and the addition of water will be followed. In addition, written records will be maintained of the battery performance. Station batteries will deteriorate with time, but precipitous failure is unlikely. The type of surveillance called for in this specification is that which has been demonstrated through experience to provide an indication of a cell becoming irregular or unserviceable long before it becomes a failure.

The equalizing charge, as recommended by the manufacturer, is vital to maintaining the ampere-hour capacity of the battery. As a check upon the effectiveness of the equalizer charge, the battery will be loaded and the voltage monitored as a function of time. If a cell has deteriorated or if a connection is loose, the voltage under load will drop excessively, indicating need for replacement or maintenance.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 15 TO

PROVISIONAL OPERATING LICENSE NO. DPR-16

JERSEY CENTRAL POWER AND LIGHT COMPANY

OYSTER CREEK NUCLEAR GENERATING STATION

Introduction

By letter dated December 23, 1975, Jersey Central Power and Light Company (JCP&L) submitted a reevaluation of emergency core cooling system (ECCS) performance for the Oyster Creek Nuclear Generating Station (OCNGS) and requested an amendment to Provisional Operating License No. DPR-16 to implement the results of the reevaluation. The ECCS reevaluation is based on (1) a revised single failure analysis, and (2) design modifications to the ECCS system as described in JCP&L's submittals of June 24, 1975 and July 15, 1975 and supplemented by letters dated November 7, 1975 and January 16, 1976. The revised single failure analysis and proposed ECCS modifications were required by conditions added to the Oyster Creek Provisional Operating License by Amendment No. 8 dated May 24, 1975. By letter dated February 5, 1976, JCP&L provided additional information requested by us regarding the ECCS reevaluation.

By letter dated January 27, 1976, JCP&L requested an amendment to the Oyster Creek license that would modify the Technical Specifications to include limiting conditions for operation (LCO) and surveillance requirements for the time delay relays that control the loading of the emergency diesel generators under postulated loss-of-coolant accident conditions. This proposed license amendment is in response to our request dated December 15, 1975.

Our evaluation of JCP&L's proposed ECCS modifications and a discussion of the need for LCO and surveillance requirements for the time delay relays are addressed in the safety evaluation dated January 21, 1976 which was issued in support of Amendment No. 12 to the Oyster Creek Provisional Operating License.

Evaluation

1.0 ECCS Reevaluation

The ECCS reevaluation submitted by letter dated December 23, 1975, is based on the results of JCP&L's revised analysis of potential single failures in the Oyster Creek Nuclear Generating Station (OCNGS). The new analysis reflects the potential for loss of both Emergency Condenser (EC) systems or failure of one EC and one Automatic Depressurization System (ADS) valve (one EC failure due to break location at the point where the EC system enters the recirculation loop, and the other EC or the ADS valve failure due to an assumed single failure). The former analysis only recognized the potential for failure of one EC or one ADS valve.

This evaluation addresses (1) the changes in the previously approved analysis, and (2) the resulting Technical Specification changes made necessary by the revised single failure assumptions and ECCS reevaluation. Our previous evaluation of JCP&L's analysis of the Oyster Creek emergency core cooling system performance was addressed in the safety evaluation issued on May 24, 1975 in support of Amendment No. 8 to the license.

1.1 Limiting Break

As before, the 0.5 ft^2 break in a recirculation line is limiting for ECCS purposes instead of the largest (4.69 ft^2) break because of the highly conservative nature of the heat transfer model during blowdown (no credit is taken for flow coastdown) as explained in our previous safety evaluation dated May 24, 1975. Consequently, the limiting single failures are of systems which are primarily designed to mitigate the consequences of small and intermediate size breaks.

1.2 Single Failure Analysis

We have reviewed JCP&L's single failure assumptions and agree that the single failures which could cause the highest calculated peak cladding temperatures following a postulated LOCA, once proper credit is taken for to design modifications, authorized by our letter dated January 21, 1976, are the loss of both emergency condensers or the loss of one emergency condenser plus the loss of one ADS valve (See the discussion above for the explanation of the dual failures).

1.3 ECCS Re-Analysis Results

Analyses for the limiting 0.5 ft^2 break in a recirculation line with these single failure assumptions predict a peak cladding temperature of 2200°F, a local cladding oxidation of less than 16.9%, and a core wide metal water reaction of less than 0.85% when the core is operated in conformance with (1) the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limits of Figure 3.10.1 of JCP&L's December 23, 1975 submittal, and (2) the power-void relationship ("B" values) stated in Section 3.10.C of the Oyster Creek Technical Specifications. It should be noted that the analysis considered various times in core life for the seven fuel types in the core. The above cited values do not all occur simultaneously.

It should also be noted that the present analysis assumes more severe single failures than the previous analysis, and the resulting MAPLHGR limits due to this change are about 1.0% lower than previously. However, for those cases at higher exposures where fuel and cladding failure was predicted in the previous analysis, Exxon Nuclear Company (ENC) has subsequently performed a more realistic but still conservative calculation which results in a more than compensating MAPLHGR limit increase. Instead of the design limit Linear Heat Generation Rate (LHGR) which ENC assumed previously to calculate fuel pellet gas release, they now have assumed values closer to (but still above) the actual maximum LHGR for each individual fuel rod. This results in a lower calculated internal gas pressure, a higher predicted rupture temperature, less rods predicted to rupture, and a lower calculated Peak Clad Temperature (PCT) for a given MAPLHGR (or conversely a higher allowable MAPLHGR at the limiting PCT). In their submittal dated February 5, 1976, JCP&L has supplied the results of several example calculations for different fuel types in the exposure range where this effect occurs, and has demonstrated to our satisfaction that this calculation method is realistic but still conservative, and represents no deviations from the approved ECCS model. Therefore, this approach and the resulting higher MAPLHGR limits are acceptable.

1.4 Operation with One Idle Recirculation Loop

1.4.1 ECCS Analysis

In their submittal dated February 5, 1976, JCP&L has presented an argument that reactor operation with a single recirculation pump inoperable would not affect ECCS analyses, provided the pump is not isolated from the system, because total flow in the reactor, and thus water inventory distribution, would be unchanged. We note that the Technical Specifications on ratio of water inventory to power ("B" values in Section 3.10.C of the Technical Specifications) would continue to be applicable with one pump inoperable, thus providing assurance that the water inventory distribution would be unchanged. We also note that for non-jet-pump plants such as Oyster Creek, no credit is taken for pump coastdown in the all-loop-operable ECCS analysis; therefore, the lesser coastdown flow actually present for four loop operation is still conservatively bounded by the five loop analysis. We find acceptable the JCP&L argument that ECCS calculations would not be changed for reactor operation with one loop inoperable.

1.4.2 Transient Analyses

JCP&L has presented an argument that operation with one recirculation pump inoperable does not alter transient analysis results because total flow, core voids, inventory, etc. are maintained the same as for operation with all loops in service. We find this acceptable.

1.4.3 Improper Idle Loop Startup Transient

JCP&L has presented an analysis assuming an idle loop containing ambient condition water (100°F) is improperly started in the fastest possible way, resulting in rapid injection of 100°F water into the core. The initial conditions assumed for the transient were conservatively at a Minimum Critical Power Ratio (MCPR) below the allowable operating MCPR limit. The calculated transient MCPR did not fall below the safety limit MCPR, thus demonstrating that idle loop startup is not the limiting transient. We find this analysis acceptable.

1.4.4 Conclusions Regarding "Partial Loop" Operation

The NRC staff has reviewed the above "partial loop" analyses and finds them acceptable. Reactor operation with one recirculation pump inoperable, but not isolated from the system, is acceptable. Accordingly, we have deleted license condition 3.E which prohibits operation of the reactor with one or more recirculation loops out of service. We have added Technical Specification 3.3.F which states that operation is permitted with one inoperable recirculation loop provided that the inoperable loop is not isolated from the reactor system.

1.5 Allowable Outage Times for Automatic Depressurization System (ADS) Valves

The licensee has proposed an allowable outage time of seven days if one ADS valve is out of service. The proposal apparently is based on the allowable outage times currently permitted in some other Boiling Water Reactors (BWRs). However, those BWR's have a High Pressure Coolant Injection (HPCI) system which can mitigate the consequences of a small break LOCA without any dependence on the ADS. No analyses have been provided for OCNGS to show that the emergency condenser systems alone can perform the same function as the HPCI system in the more recently designed BWR's.

The situation most analogous to OCNGS for small breaks appears to be a Pressurized Water Reactor (PWR) for which we permit one of two high head safety injection pumps to be out of service for three days. Accordingly, to be consistent with the considerations leading to that allowable outage time for PWR's, we have modified Section

3.4.B.2 of the Technical Specifications to permit one ADS to be out of service for three days provided that the motor operated isolation valves and condensate makeup valves in both isolation condensers are demonstrated to be operable daily. In addition, we have modified Section 3.4.B.3 of the Technical Specifications to indicate that the reactor pressure shall be reduced to 110 psig or less within 24 hours if Specifications 3.4.B.1 and 3.4.B.2 are not met. The Technical Specifications previously did not specify a time limit for reducing reactor pressure.

The above mentioned changes to the Technical Specifications have been discussed with and concurred in by the licensee.

1.6 Technical Specifications for Time Delay Relays

We have reviewed JCP&L's January 27, 1976 request for a license amendment to provide limiting conditions for operation and surveillance requirements for the time delay relays that control the loading of the emergency diesel generators under postulated LOCA conditions. JCP&L has analyzed the sequencing requirements for the auxiliary systems and emergency core cooling systems. The analysis included an evaluation of the maximum acceptable tolerances for the diesel loading times. Based on our review, we find that the sequencing times, timer tolerances, and surveillance frequency are acceptable.

1.7 ECCS Modifications

The licensee has stated that the proposed modifications with regard to the electrical supply for the emergency core cooling system will be installed during the current outage and prior to resuming power operation. We find this schedule acceptable. Based on our review of (1) JCP&L's single failure analysis of the electrical systems associated with ECCS performance, (2) the proposed ECCS modifications, and (3) the reevaluation of ECCS performance, we find that JCP&L has fulfilled license condition 3.F previously added to the license on May 24, 1975 by Amendment No. 8. Accordingly, we are deleting condition 3.F from the license.

1.8 Conclusion

We conclude that operation of the reactor in accordance with the Technical Specifications will meet the requirements of 10 CFR Part 50, Section 50.46.

Environmental Conditions

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

Conclusion

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

Dated: February 24, 1976

20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 24th day of February, 1976.

FOR THE NUCLEAR REGULATORY COMMISSION

/S/

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

KRG

OFFICE	ORB#3 cp	ORB#3	OELD	ORB#3	RL:AD/ORs	
SURNAME	CParrish:kmf	WPaulson		GLear	KRGoller	
DATE	2/ 20 /76	2/ 20 /76	2/ 23 /76	2/ 23 /76	2/ 24 /76	

published in the FEDERAL REGISTER on January 16, 1976 (41 FR 2448). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental statement, negative declaration or environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the applications for amendment dated December 23, 1975, (and supplement dated February 5, 1976), and January 27, 1976, (2) Amendment No. 15 to License No. DPR-16, (3) the Commission's related Safety Evaluation, and (4) the Commission's Safety Evaluation dated January 21, 1976 which evaluated the ECCS modification and was issued in support of Amendment No. 12 to the license. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Ocean County Library, 15 Hooper Avenue, Toms River, New Jersey.

A copy of items (2), (3) and (4) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D. C.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-219

JERSEY CENTRAL POWER & LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENT

TO PROVISIONAL OPERATING LICENSE

Notice is hereby given that the U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 15 to Facility Operating License No. DPR-16 issued to Jersey Central Power & Light Company which revised Technical Specifications for operation of the Oyster Creek Nuclear Generating Station, located in Ocean County, New Jersey. The amendment is effective as of its date of issuance.

In accordance with the licensee's applications for amendment dated December 23, 1975 (as supplemented) and January 27, 1976, the amendment would (1) revise the provisions in the Technical Specifications relating to operating limits (maximum average planar linear heat generation rate) and operability requirements of the automatic depressurization system based on a reevaluation of the Oyster Creek Nuclear Generating Station ECCS performance, and (2) add limiting conditions for operation and surveillance requirements for the time delay relays which control the loading of the diesel generators under postulated loss-of-coolant accident conditions.

The applications for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Notice of Proposed Issuance of Amendment to Provisional Operating License in connection with this action was published in the FEDERAL REGISTER on January 16, 1976 (41 FR 2448). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental statement, negative declaration or environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the applications for amendment dated December 23, 1975, (and supplement dated February 5, 1976), and January 27, 1976, (2) Amendment No. 15 to License No. DPR-16, (3) the Commission's related Safety Evaluation, and (4) the Commission's Safety Evaluation dated January 21, 1976 which evaluated the ECCS modification and was issued in support of Amendment No. 12 to the license. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Ocean County Library, 15 Hooper Avenue, Toms River, New Jersey.

A copy of items (2), (3) and (4) 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 24th day of February, 1976.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

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