FEB 1 9 1976

Docket No. 50-219

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

Gentlemen:

The Commission has issued the enclosed Amendment No. 13 to Provisional Operating License No. DPR-16 for Unit 1 of the Oyster Creek Nuclear Generating Station. This amendment consists of changes to the Technical Specifications and are based on our letters to you dated September 25, 1975 and December 17, 1975.

This amendment revises the Technical Specifications to (1) add requirements that would limit the period of time operation can be continued with immovable control rods that could have control rod drive mechanism collet housing failures and (2) require increased control rod surveillance when the possibility of a control rod drive mechanism collet housing failure exists.

We have evaluated the potential for environmental impact of plant operation in accordance with the enclosed amendment and 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. We have also concluded that there is reasonable assurance that the health and safety of the public will not be endangered by this action.

OFFICE ➤						
SURNAME ➤						
DATE ➤						

A copy of the related Federal Register Notice is also enclosed. Our Safety Evaluation relating to this action was forwarded to you with our letter dated September 25, 1975.

Sincerely,

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

Enclosures:
1. Amendment No. 13
2. Federal Register Notice

cc w/enclosures:
See next page

DISTRIBUTION:
NRC PDR
Local PDR
Docket
ORB#3 Rdg
VStello
KRGollerTJCarter
CParrish
WPaulson
OELD
OI&E (7)
BJones (4)
BScharf (10)
JMcGough
ACRS (16)
OPA (Clare Miles)
TBAbernathy
JRBucahanan
Gray files
extra cps

OFFICE➤	ORB#3 cp	ORB#3	OELD DL	ORB#3	RL:AD/ORs KRG	
SURNAME➤	CParrish:kmf	WPaulson	DSwanson	GLear G	KRGoller	
DATE➤	2/11/76	2/11/76	2/13/76	2/17/76	2/17/76	

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

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
NUC[illegible]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. 11
License No. DPR-16

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

   A. 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;

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

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

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

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

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

FOR THE NUCLEAR REGULATORY COMMISSION

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 19, 1976

<u>ATTACHMENT TO LICENSE AMENDMENT</u>

<u>CHANGE TO THE TECHNICAL SPECIFICATIONS</u>

<u>PROVISIONAL OPERATING LICENSE NO. DPR-16</u>

<u>DOCKET NO. 50-219</u>

Delete existing pages 3.2-2, 3.2-6 and 4.2-1 of the Technical Specifications and insert the attached revised pages. The changed areas on the revised pages are shown by marginal lines.

Any four rod group may contain a control rod which is valved out of service provided the above requirements and Specification 3.2.A are met. Time zero shall be taken as the de-energization of the pilot scram valve solenoids.

4. Control rods which cannot be moved with control rod drive pressure shall be considered inoperable. If a partially or fully withdrawn control rod drive cannot be moved with drive or scram pressure the reactor shall be brought to a shutdown condition within 48 hours unless investigation demonstrates that the cause of the failure is not due to a failed control rod drive mechanism collet housing. Inoperable control rods shall be valved out of service, in such positions that Specification 3.2.A is met. In no case shall the number of rods valved out of service be greater than six during power operation. If this specification is not met, the reactor shall be placed in the shutdown condition.

5. Control rods shall not be withdrawn for approach to criticality unless at least three source range channels have an observed count rate equal to or greater than 3 counts per second.

6. The control rod density shall be greater than 3.5 percent during all modes of reactor operation.

C. Standby Liquid Control System

1. The standby liquid control system shall be operable at all times when the reactor is not shutdown by the control rods such that Specification 3.2.A is met and except as provided in Specification 3.2.C.3.

2. The standby liquid control solution shall be maintained within the volume - concentration requirement area in Figure 3.2-1 and at a temperature not less than the temperature presented in Figure 3.2-2 at all times when the standby liquid control system is required to be operable.

3. If one standby liquid control system pumping circuit becomes inoperable during the run mode and Specification 3.2.A is met the reactor may remain in operation for a period not to exceed 7 days, provided the pump in the other circuit is demonstrated daily to be operable.

D. Reactivity Anomalies

The difference between an observed and predicted control rod inventory shall not exceed the equivalent of one percent in reactivity. If this limit is exceeded and the discrepancy cannot be explained, the reactor shall be brought to the cold, shutdown condition by normal orderly shutdown procedure. Operation shall not be permitted until the cause has been evaluated and appropriate corrective action has been completed. The NRC shall be notified within 24 hours of this situation in accordance with Specification 6.6.B.

Bases:

Limiting conditions of operation on core reactivity and the reactivity control systems are required to assure that the excess reactivity of the reactor core is controlled at all times. The conditions specified herein assure the capability to provide reactor shutdown from steady state and transient conditions and be many more than the six allowed by the specification, particularly late in the operating cycle; however, the occurrence of more than six could be indicative of a generic problem and the reactor will be shutdown. Also if damage within the control rod drive mechanism and in particular, cracks in drive internal housings, cannot be ruled out, then a generic problem affecting a number of drives cannot be ruled out. Circumferential cracks resulting from stess assisted intergranular corrosion have occurred in the collet housing of drives at several BWRs. This type of cracking could occur in a number of drives and if the cracks propagated until severance of the collet housing occurred, scram could be prevented in the affected rods. Limiting the period of operation with a potentially severed collet housing and requiring increased surveillance after detecting one stuck rod will assure that the reactor will not be operated with a large number of rods with failed collet housings. Placing the reactor in the shutdown condition inserts the control rods and accomplishes the objective of the specifications on control rod operability. This operation is normally expected to be accomplished within eight hours.

The source range monitor (SRM) system[9] performs no automatic safety function. It does provide the operator with a visual indication of neutron level which is needed for knowledgeable and efficient reactor startup at low neutron levels. The results of the reactivity accidents are functions of the initial neutron flux. The requirement of at least 3 cps assures that any transient begins at or above the initial value of 10^{-8} of rated power used in the analyses of transients from cold conditions. One operable SRM channel would be adequate to monitor the approach to critical using homogeneous patterns of scattered control rods. A minimum of three operable SRM's is required as an added conservatism.

The standby liquid control system is designed to bring the reactor to a cold shutdown condition from the full power steady state operating condition at any time in core life independent of the control rod system capabilities[10]. If the reactor is shutdown by the control rod system and would be subcritical in its most reactive condition as required in Specification 3.2.A, there is no requirement for operability of this system. To bring the reactor from full power to cold shutdown sufficient liquid control must be inserted to give a negative reactivity worth equal to the combined effects of rated coolant voids, fuel Doppler, xenon, samarium, and temperature change plus shutdown margin. This requires a boron concentration of 600 ppm in the reactor. An additional 25% boron, which results in an average boron concentration in the reactor of 750 ppm, is inserted to provide margin for mixing uncertainties in the reactor. The system is required to insert the solution in a time interval between 60-120 minutes to provide for good mixing in the reactor and to override the rate of reactivity insertion due to cooldown of the reactor following the xenon peak.

The liquid control tank volume-concentration requirements of Figure 3.2-1 assure that the above requirements for liquid control insertion are met with one 30 gpm liquid control pump. The point (1937 gal, 19.6%)[11] results in the required amount of solution being inserted into the reactor is not less than 60 minutes, and therefore, defines the maximum concentration-minimum volume requirement. The point (3737 gal, 10.5%)[11], results in the required amount of solution being injected into the reactor is

4.2 REACTIVITY CONTROL

Applicability: Applies to the surveillance requirements for reactivity control.

Objective: To verify the capability for controlling reactivity.

Specification: A. Sufficient control rods shall be withdrawn following a refueling outage when core alterations were performed to demonstrate with a margin of 0.25% Δk that the core can be made subcritical at any time in the subsequent fuel cycle with the strongest operable control rod fully withdrawn and all other operable rods fully inserted.

B. The control rod drive housing support system shall be inspected after reassembly.

C. 1. After each major refueling outage and prior to resuming power operation, all operable control rods shall be scram time tested from the fully withdrawn position with reactor pressure above 800 psig.

2. Following each reactor scram from rated pressure, the mean 90% insertion time shall be determined for eight selected rods. If the mean 90% insertion time of the selected control rod drives does not fall within the range of 2.4 to 3.1 seconds or the measured scram time of any one drive for 90% insertion does not fall within the range of 1.9 to 3.6 seconds, an evaluation shall be made to provide reasonable assurance that proper control rod drive performance is maintained.

3. Following any outage not initiated by a reactor scram, eight rods shall be scram tested with reactor pressure above 800 psig provided these have not been measured in six months. The same criteria of 4.2.C.(2) shall apply.

D. Each partially or fully withdrawn control rod shall be exercised at least once each week. This test shall be performed at least once per 24 hours in the event power operation is continuing with two or more inoperable control rods or in the event power operation is continuing with one fully or partially withdrawn rod which cannot be moved and for which control rod drive mechanism damage has not been ruled out. The surveillance need not be completed within 24 hours if the number of inoperable rods has been reduced to less than two and if it has been demonstrated that control rod drive mechanism collet housing failure is not the cause of an immovable control rod.

E. Surveillance of the standby liquid control system shall be as follows:

1.	Pump operability	Once/month
2.	Boron concentration determination	Once/month
3.	Functional test	Each refueling outage
4.	Solution volume and temperature check	Once/day

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. 13 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.

This amendment revises the Technical Specifications to (1) add requirements that would limit the period of time operation can be continued with immovable control rods that could have control rod drive mechanism collet housing failures and (2) require increased control rod surveillance when the possibility of a control rod drive mechanism collet housing failure exists.

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. Notice of the Proposed Issuance of Amendment to Facility Operating Licenses in connection with this action was published in the FEDERAL REGISTER on January 8, 1976 (41 F.R. 1548). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

OFFICE➤	ORB#3 CP	ORB#3	OELD	ORB#3		
SURNAME➤	CParrish:kmf	WPaulson	DSwanson	GLear GL		
DATE➤	2/ 11 /76	2/ 11 /76	2/ 13 /76	2/ 17 /76		

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

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 Commission's letters to Jersey Central Power and Light Company dated September 25, 1975, and December 17, 1975, (2) Amendment No. 13 to License No. DPR-16, and (3) the Commission's related Safety Evaluation issued on September 25, 1975. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W. Washington, D. C. and at the Ocean County Library, 15 Hooper Avenue, Toms River, New Jersey 08753.

A single copy of items (1) through (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 19th day of February, 1976

FOR THE NUCLEAR REGULATORY COMMISSION

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

OFFICE ➤						
SURNAME ➤						
DATE ➤						

Trans w/letter dtd. 9-25-75

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT TO LICENSE NO. DPR-16

AND

CHANGES TO THE TECHNICAL SPECIFICATIONS

INOPERABLE CONTROL ROD LIMITATIONS

JERSEY CENTRAL POWER AND LIGHT COMPANY

OYSTER CREEK NUCLEAR GENERATING STATION UNIT 1

DOCKET NO. 50-219

INTRODUCTION

On June 27, 1975, Commonwealth Edison Company (CE) informed NRC that cracks had been discovered on the outside surface of the collet housings of four control rod drives at Dresden Unit 3[1]. The cracks were discovered while performing maintenance of the control rod drives; the reactor was shutdown for refueling and maintenance. In a letter dated July 3, 1975, CE informed us that if the cracks propagated until the collet housing failed, the affected control rod could not be moved[2]. In a meeting with representatives of General Electric (GE) and CE the NRC staff was advised that further inspections revealed cracks in 19 of the 52 Dresden 3 control rod drives inspected, in one spare Dresden 2 control rod drive, in one Vermont Yankee spare control rod drive and in two GE test drives[3]. In a report dated July 30, 1975, after additional rod drives were inspected, CE stated that cracks had been found in 24 of 65 drives inspected[4]. Recently, the Tennessee Valley Authority reported that cracks were found in the collet housing of

---

(1) Telegram to J. Keppler, Region III of the NRC, June 27, 1975, Docket No. 50-249.

(2) Letter from B. B. Stephenson, Commonwealth Edison Company to James G. Keppler, U. S. Nuclear Regulatory Commission, July 3, 1975, Docket No. 50-249.

(3) Memo from L. N. Olshan, Division of Technical Review (DTR) to T. M. Novak, DTR, "Meeting on Cracks Found in Dresden 3 Control Rod Drive Collet Retainer Tubes," July 18, 1975.

(4) Letter from B. B. Stephenson, Commonwealth Edison Company to James G. Keppler, U. S. Nuclear Regulatory Commission, July 30, 1975, Docket No. 50-249.

seven of nineteen drives inspected at Browns Ferry 1 and Vermont Yankee found cracks in the collet housing of 4 of 10 control rod drives inspected. Because a number of control rod drives have been affected, because complete failure of the drive collet housing could prevent scram of the affected rod, and because we do not consider existing license requirements adequate in view of the collet housing cracks experienced, we have concluded that the Technical Specifications should be changed for those reactors with control rod drive designs susceptible to collet housing cracks. The change should assure that reactors which could be affected would not be operated for extended periods of time with a control rod which cannot be moved.

DESCRIPTION

The control rod drive is a hydraulically operated unit made up primarily of pistons, cylinders and a locking mechanism to hold the movable part of the drive at the desired position. The movable part of the drive includes an index tube with circumferential grooves located six inches apart. The collet assembly which serves as the index tube locking mechanism contains fingers which engage a groove in the index tube when the drive is locked in position. In addition to the collet, the collet assembly includes a return spring, a guide cap, a collet retainer tube (collet housing) and collet piston seals. The collet housing surrounds the collet and spring assembly. The collet housing is a cylinder with an upper section of wall thickness 0.1 inches and a lower section with a wall thickness of about 0.3 inches. The cracks occurred on the outer surface of the upper thin walled section near the change in wall thickness.

1. Consequences of Cracking

The lower edges of the grooves in the index tube are tapered, allowing index tube insertion without mechanically opening the collet fingers, as they can easily spring outward. If the collet housing were to fail completely at the reported crack location, the coil collet spring could force the upper part of the collet housing and spring retainer upward, to a location where the spring and spring retainer would be adjacent to the collet fingers. The clearance between the collet fingers and the spring when in this location will not permit the collet fingers to spring out of the index tube groove. This would lock the index tube in this position so that the control rod could not be inserted or withdrawn.

The failure of up to six control rods to operate has previously been evaluated and the Technical Specifications presently allow up to six rods to be inoperable. If more than six rods are inoperable or if the scram reactivity rate is too small or if shutdown reactivity requirements are not met, the existing Technical Specifications require the reactor to be brought to a cold shutdown condition. Reactor power operation with up to six rods inoperable would not involve a new hazards consideration nor would it endanger the health and safety of the public.

2. Probable Cause of Cracking

The cause of the cracking appears to be a combination of thermal cycling and intergranular stress corrosion cracking. The thermal cycling results from insertion and scram movements. During these movements hot reactor water is forced down along the outside of the collet housing, while cool water is flowing up the inside and out of flow holes in the housing. These thermal cycles are severe enough to yield the material, leaving a high residual tensile stress on the outer surface.

The collet housing material is type 304 austenitic stainless steel. The lower portion of the collet housing has a thicker wall and its inner surface is nitrided for wear resistance. In 1960-61, similar drives using high hardness 17-4 PH material for index tubes and other parts were found to have developed cracks. The problem caused GE to switch to nitrided stainless steel. The nitriding process involves a heat treatment in the 1050 F to 1100 F range, which sensitizes the entire collet housing, making it susceptible to oxygen stress corrosion cracking.

The cooling water used in the drives is aerated water. This water contains sufficient oxygen for stress corrosion to occur in the sensitized material if it is subjected to the proper combination of high stresses and elevated temperatures.

We believe that the cracking is caused by a combination of thermal fatigue and stress corrosion. GE has determined that both full stroke insertion and scram will cause high thermal stress. The cracks are completely intergranular and extensively branched, indicating that corrosion is a major factor. The type of thermal cycling, plus the buildup of corrosion products in the cracks between cycles probably results in a ratcheting action. This is also indicated by the "bulged" appearance of the cracks on the OD.

3. Probability of Early Failure

We believe that the cracking is progressive and is cycle dependent. Although the details of the cracking process are still not clear, we have not identified any mechanism that would cause rapid cracking with progression to complete circumferential failure.

The axial loads on the housings are very low at all times so that through wall cracks would have to progress at least 90% around the circumference before there would be concern about a circumferential failure. Although one housing at Dresden 3 had three cracks which nearly joined around the circumference, no cracks at Dresden 3 were through wall and none of the housings examined approached the degree of cracking necessary for failure. The collet housing has three flow holes in the thin section equally spaced around the circumference. The observed cracks have been confined primarily to the areas below and between the holes and near the area where the wall thickness of the collet housing changes. Since all the cracks except those located at the change in wall thickness are fairly shallow and since those at the change in wall thickness are largely confined to the circumferential area between holes, the net strength of the cracked housings is still far greater than necessary to perform their function.

A test drive at GE that had experienced over 4000 scram cycles had a more extensive developed crack pattern. Although the satisfactory experience with this cracked test housing is encouraging, its performance may not be correlated directly to that of drives in service, as this test drive was subjected to lower temperatures, and possibly less severe thermal cycles than could be encountered in actual service. The cracks were first noticed on the test drive after about 2000 cycles* - many more cycles than the cracked housings at Dresden 3 had experienced.

The chance that a large number of collet housing would fail completely at about the same time is very remote. This is primarily true because the distributions of failures by cracking mechanisms such as stress corrosion and fatigue are not linear functions. That is, failure is a function of log time or log cycles. Distribution of failures of similar specimens generally follow a log normal pattern, with one to two orders of magnitude in time or cycles between failures of the first and failures of the last specimen. As no collet housing has yet failed, we are confident that there would be very few, if any, failures during the next time period corresponding to the total service life to date.

4. Changes to Technical Specifications

Existing limiting conditions of operation allow operation to continue with up to six inoperable control rods. Existing surveillance requirements specify that daily surveillance of the condition of all fully or partially withdrawn rods would not have to begin until two rods are found inoperable. We do not consider that these existing limiting conditions of operation and surveillance requirements sufficiently limit the possibility of operating for an extended period of time with a number of rod drive mechanisms which cannot be moved. We have therefore concluded that the Technical Specifications should be changed as discussed below.

(a) One stuck control rod does not create a significant safety concern. However, if a rod cannot be moved and the cause of the failure cannot be determined, the rod could have a failed collet housing. A potentially failed collet housing would be indicative of a problem which could eventually affect the scram capability of more than one control rod. Since the cracks appear to be of a type which propagate slowly, it is highly unlikely that a second control rod would experience a failed collet housing within a short period of time after the first failure. Therefore, a period of time of 48 hours can be allowed to determine the cause of failure. This period is considered long enough to determine if the cause of failure is not in the drive mechanism, yet short enough to be reasonably assured that a second collet failure does not occur. Therefore Section 3.2.B.4 should be expanded to require that if a control rod cannot be moved during normal operation, testing or scram, the reactor shall be shutdown within 48 hours if the reason that it cannot be moved cannot be shown to be due to causes other than a failed collet housing.

(b) If a control rod drive cannot be moved, the cause of the stuck rod might be a problem affecting other rods. To ensure prompt detection of any additional control rod drive failures which could prevent movement, Section 4.2.D should be expanded to require surveillance every 24 hours of all partially and fully withdrawn rods if one rod drive is found to be stuck.

Until permanent corrective measures are taken to resolve the potential for stuck control rods due to failed collet housings, we believe that these additional specifications provide reasonable assurance that an unacceptable number of control rod collet housing will not fail during operation. Upon completion of the investigations being performed by GE, additional corrective actions may permit revision of these requirements.

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.

Date: SEP 2 5 1975