U. S. ATOMIC ENERGY COMMISSION
DIRECTORATE OF REGULATORY OPERATIONS
REGION I

RO Inspection Report No. 50-219/72-03

Subject: Jersey Central Power & Light Company

Oyster Creek

License No. DPR-16

Location: Forked River, New Jersey

Priority

Category C

Type of Licensee: 1930 MWt, BWR

Type of Inspection: Special (to observe the scheduled turbine trip test & reactor startup), Announced

Dates of Inspection: April 21, 1972

Dates of Previous Inspection: February 23, 24, 25, 29 & March 1, 1972

Principal Inspector: F. S. Cantrell 5/24/72
F. S. Cantrell, Reactor Inspector Date

Accompanying Inspectors: None
Date

Date

Other Accompanying Personnel: None
Date

Reviewed by: R. L. Spessard 5/24/72
R. L. Spessard, Reactor Inspector Date

Date

Proprietary Information: None

Section I

Enforcement Action: None

Licensee Action on Previously Identified Enforcement Matters: Not applicable.

Unresolved Items:

The extent of corrective action concerning the cracks observed in two safety valves. (Paragraph 4)

Status of Previously Reported Unresolved Items: Not applicable.

Unusual Occurrences:

A. Cracks were found on the seat bushing of two safety valves that were removed from the main steam lines during the September - November 1971 outage. (Paragraph 4)

B. The outage scheduled to begin April 21, 1972 was postponed for about one week due to labor problems. (Paragraph 3)

C. The containment atmosphere which was de-inerted in anticipation of the scheduled outage was re-inerted within 24 hours. (Paragraph 2)

Persons Contacted:

Mr. T. J. McCluskey, Station Superintendent
Mr. Don Ross, Technical Supervisor
Mr. I. R. Finfrock, Manager, Nuclear Generating Station

Management Interview:

The following subjects were discussed with Mr. McCluskey and Mr. Finfrock on April 21, 1972:

A. The inspector requested a copy of the shutdown schedule, and a copy was provided. (Paragraph 3)

B. The inspector asked if the General Office Review Board had reviewed the scheduled turbine trip test in light of finding cracks in two safety valves, and pointed out that the cracks were in the primary system pressure boundary.

   After some discussions, Mr. Finfrock agreed that the test would not be performed unless the GORB met and approved the tests in light of finding cracks in two safety valves.

The inspector was subsequently informed by telephone that the GORB met and re-approved the turbine trip test. According to Mr. Finfrock, one of the key points considered by GORB was a transient analysis that has not yet been submitted to the AEC. This analysis shows that under the most adverse conditions from 1930 MWt (approximately 100 MWt higher than the currently scheduled test), the maximum pressure expected would only be equivalent to the relief set point of the first four safety valves (1212 psi). The GORB concluded there could be no safety problem unless the safety valve actuated. (Paragraph 4)

C. The preparations for the scheduled outage in light of the existing labor negotiations were discussed. Mr. Finfrock stated that special plans had been made to staff the plant in the event the employees refused to cross the picket line. He stated that if it appeared a picket line would interfere with the shutdown schedule, the outage would be postponed for a week.

Mr. McCluskey subsequently informed the inspector by telephone that the refueling outage scheduled to start at 10:00 pm on April 21, 1972 had been postponed due to a breakdown in labor negotiations between JCP&L, the International Brotherhood of Electrical Workers (IBEW) which represents Oyster Creek's hourly employees, and the local construction trade unions. According to Mr. McCluskey, JCP&L entered into a contract with a New Jersey firm to perform the turbine overhaul using local labor under General Electric supervision. The local labor would belong to the IBEW. The local construction trade union previously objected to the use of outside GE employees for the turbine overhaul and established a picket line at the plant site.* Contrary to their understanding, JCP&L was informed on April 21, 1972 that if the construction trade unions established a picket line around the plant, the IBEW employees would honor the picket line. Information available to JCP&L indicated a picket line would be established on April 24, 1972 if the outage had commenced.

*Previously reported in Inquiry Report 50-219/71-07.

Section II

Det..ils oi Subjects Discussed in Section I

1. General

Oyster Creek was required to perform a turbine trip test from full power as a condition of the authorization to operate at 1930 Mwt. Due to problems with the reheater, the plant has been unable to generate steam of sufficient quality for the turbine to use all of the steam. As a result, the plant has been operated to approximately 100 Mwt below the authorized level. The General Office Review Board directed the plant staff to conduct the turbine trip test at the operating power level, at the latest, just prior to the refueling outage. A commitment was made to the Division of Reactor Licensing to this effect. The test was scheduled for 10:00 pm on April 21, 1972.

The inspector was informed on April 21 at 9:00 pm that the scheduled outage had been postponed at least one week due to the failure of JCP&L, the IBEW and the construction trade union to reach an agreement as to who would perform the scheduled turbine maintenance.

2. Containment De-inerting

In anticipation of the turbine trip test scheduled for 10:00 pm on April 21, 1972, the nitrogen in the containment atmosphere was purged to establish a normal oxygen content. The oxygen content was increased above 5% (TS limit for routine operation) at approximately 3:00 pm on April 21, 1972. Technical Specifications permit de-inerting to begin 24 hours prior to a scheduled shutdown. A decision to postpone the shutdown (for approximately one week) was made at 9:00 pm on April 21, 1972. A nitrogen delivery was scheduled for 1:00 am on April 22, 1972. Mr. McCluskey informed the inspector of the problem. After a review of this matter at the Regional level, Mr. McCluskey was informed that the intent of the Technical Specifications would be met if the containment was re-inerted (oxygen less than 5%) within 24 hours of the start of the initial purge. Mr. McCluskey informed the inspector by telephone on April 24, 1972 that the re-inerting was completed at 6:50 am on April 22, 1972.

3. Outage Schedule

The refueling outage was scheduled for a total of 32 days. Major items scheduled include: sipping the fuel to determine which assemblies include leaking pins; replacement of 136 fuel assemblies (132 GE plus 4 Jersey Nuclear); replacement of the four control rod drives that settled at notch "02" on the April 13, 1972 scram*; replacement of the specimen holder in the reactor that was removed during the September - November 1971 outage; modifying the operator for the main steam isolation valves as recommended by the manufacturer (remove the cushion spud**; inspect reactor internals; and inspect turbine generator.

4. Cracks in Safety Valve Bushings

Mr. McCluskey stated that during the September - November 1971 outage, five safety valves were replaced with five clean tested spare valves. The plans were to test the five valves removed using nitrogen; however, a correlation between testing with cold nitrogen and hot steam was not available. As a result, it was necessary to send the valves to the manufacturer's shop for testing and to determine the correlation between cold nitrogen and hot steam for future testing. Efforts to decontaminate the valves to suitable levels for shipment to the manufacturer's shop (less than 2 mR/hr) were unsuccessful until the valve seat bushing was unscrewed from the valve body. When initial decontamination efforts on the seat bushing of the first valve were unsuccessful, a dye check showed radial cracks on the seat and a circumferential crack approximately 4.4 inches from the base, at a point where the wall thickness completed the transition from 1.4 inches to 0.75 inches. It was necessary to grind to a maximum depth of 0.12 inches to remove the circumferential crack. (Attachment 1)

Without any further attempt to decontaminate, the seat bushing was removed from the second valve and was dye checked. Cracks were detected at the same locations as in the first valve examined. In addition, several vertical cracks about 1/2 inch long were noted about ten inches above the base (the point at which water could have been standing if the valves were cold).

The remaining three valves were disassembled and dye checked but did not show the crack indications found on the first two valves, according to Mr. McCluskey.

The valves are Dresser "Maxiflow Safety Valves", Model 6-3777QA, with a six inch inlet and an eight inch outlet (Attachment 2). The seat bushing is ASTM A182, Grade F304 stainless steel. The base (or valve housing) is ASTM A216, Grade WCA carbon steel. Oyster Creek has 16 safety valves installed in the primary system, and five spare valves.

The valve seat of the second valve was shipped to General Electric, San Jose, California by air freight for metallurgical analysis on April 20, 1972. The results are currently being evaluated by General Electric.

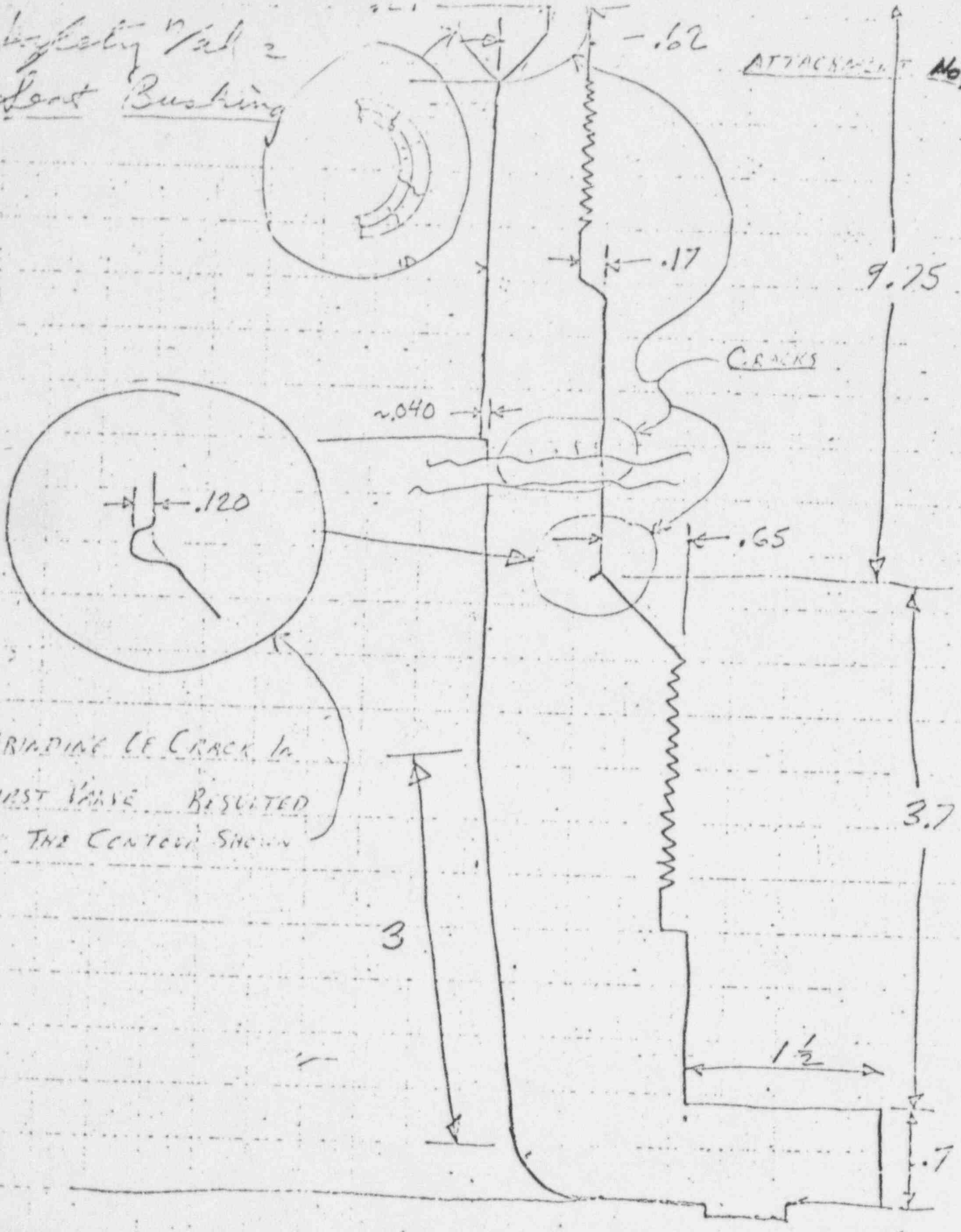
*Previously reported in Inquiry Report No. 219/72-07.
**Previously reported in Inquiry Report No. 219/71-11.

Ten safety valve repair kits were ordered from GE (all that were available) to use in replacing cracked seats.

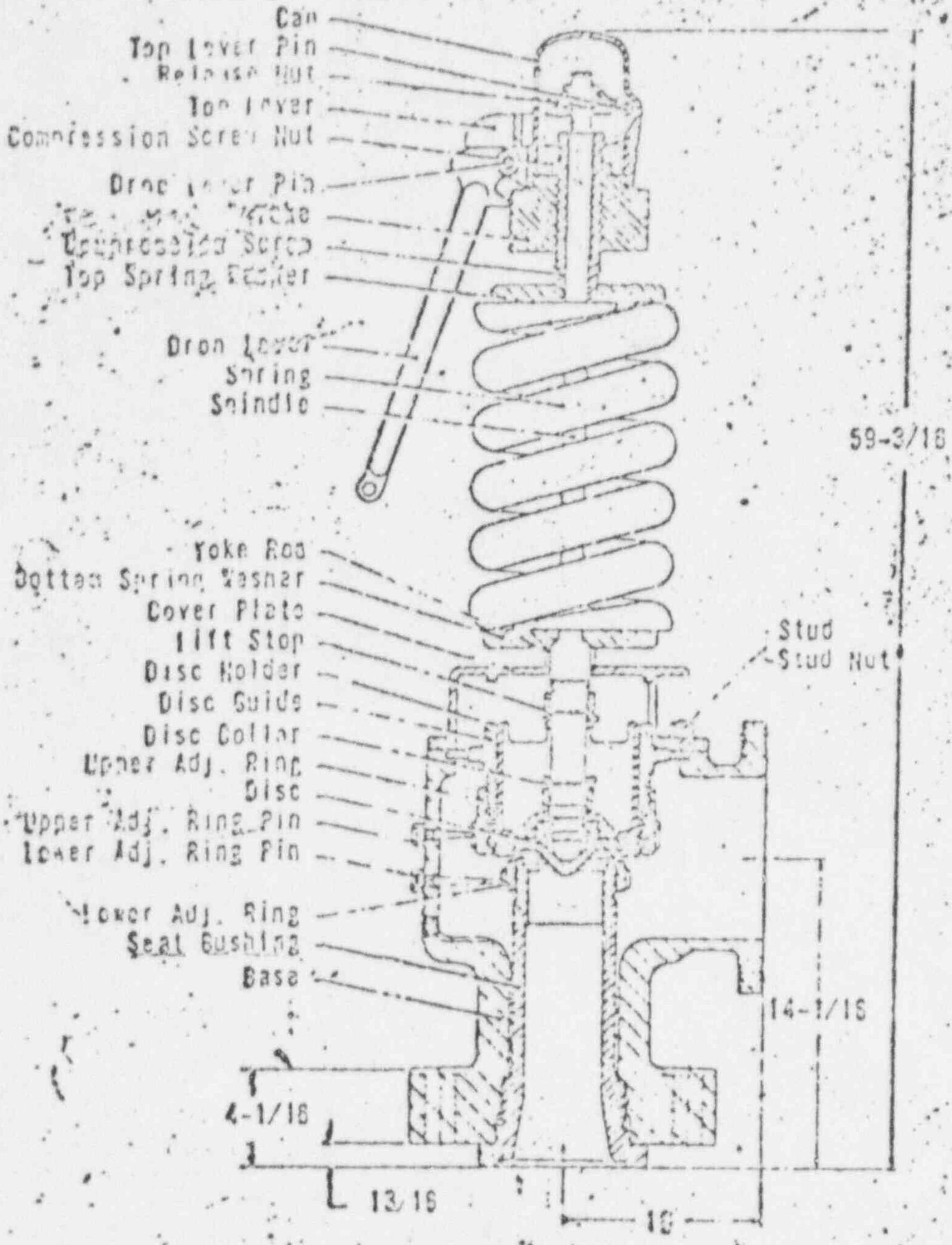
The General Office Review Board (GORB) held a special meeting on April 21, 1972 to review the scheduled turbine trip test in light of the cracks found in the safety valves. The GORB approved the test as scheduled. The basis of approval was a new transient analysis that shows that the peak pressure that would be experienced from 1930 Mwt (approximately 100 Mwt higher than current operations) would be 1212 psi (the set point of the four safety valves with the lowest set point). Mr. Finfrock reported this analysis had not been submitted to the Commission

ATTACHMENT No 1

Safety Valve 2
Seat Bushing



ATTACHMENT No2



INLET - 6" - 1500# SPECIAL TONGUE (PER G.E. DRG. 159A2825)

OUTLET - 8" - 150# ASA STANDARD RAISED FACE

TSURU

NUCLEAR

MILLSTONE

DRESDEN 2

QUAD CITIES

GENERAL E
APED

VPF# 1 9

EP#

Orifice Area

Approximate

Maximum Press

Maximum Temp

Certified:

APR 1 9 1972 3944

J. G. Keppler, Chief, Reactor Testing & Operations Br.
Division of Compliance, HQ

CO INQUIRY REPORT NO. 50-219/72-08
JERSEY CENTRAL POWER & LIGHT COMPANY
OYSTER CREEK - BWR
POSSIBLE TECHNICAL SPECIFICATION VIOLATION - HIGH STACK RELEASE RATE

The subject inquiry report is forwarded for action. We recommend that the intent of the Specification on stack release rate (paragraph 3.6.A) be discussed with DRL to determine how this specification should be inspected. Is the TS limit of $0.21/\bar{E}$ as determined at equilibrium full power conditions supposed to be the limit for all conditions until a different $\bar{E}$ is determined, or can a series of determinations be made under transient conditions and the results used during subsequent transients as the release limit? Is this data transferable from reactor to reactor?

As the specification is written, it appears that the $\bar{E}$ determined once per month is the fixed limit; however, Safety Guide No. 21, dated December 29, 1971, Note 2, p. 21.4 states, "For those processes or other conditions which are changed frequently, an isotopic analysis should be done following each change until a pattern has been established which can be used to predict the isotopic composition of the reactor effluent". It should be noted that if one determines a release limit based on a sample taken to calculate $\bar{E}$ while the mechanical vacuum pump is in operation, the analysis will only tell if the TS was violated. By the time the analysis is made, 4 - 6 hours later, the release rate will have dropped by a factor of 10 - 100.

Mr. McCluskey stated that JCP&L will submit a report of this occurrence to DRL. The report will either show a TS violation or attempt to justify why the occurrence should not be considered a violation. GE told JCP&L that they had data to show that $\bar{E}$ decreases under the conditions experienced. We plan to follow the resolution of this matter by JCP&L and will keep you informed as is appropriate.

E. T. Carlson
Senior Reactor Inspector

B/384

Enclosure:
Subject Inquiry Report (18 cys)

cc: L. Kornblith, CO
R. H. Engelken, CO
CO Files

OFFICE ▶		CO			
SURNAME ▶		Cantrell:smg	Carlson		
DATE ▶		4/19/72			

Form AEC-318 (Rev. 9-53) U.S. GOVERNMENT PRINTING OFFICE 1969-O-364-598

~~8304180067~~ 1p.

CO Inquiry Report No. 50-219/72-08

Subject: Jersey Central Power & Light Company

License No.: DPR-16

Facility: Oyster Creek - BWR

Title: Possible Technical Specification Violation - High Stack Release Rate

Prepared by: ______________________ ____________

F. S. Cantrell, Jr., Reactor Inspector Date

A. Date and ... AEC was Informed:

By telephone call from Mr. T. J. McCluskey, Station Superintendent, April 15, 1972 (at home).

B. Description of Particular Event or Circumstance:

During a plant startup on April 14, 1972 (following the reactor scram reported in Inquiry Report No. 219/72-07), the stack release rate reached 330,000 uCi/second when the mechanical vacuum pumps were started and remained above 280,000 uCi/second for 20 minutes (the maximum release rate based on the $\bar{E}$ calculated prior to the scram). Technical Specifications limit the stack release rate to $0.21/\bar{E}$ Ci/sec. $\bar{E}$ at equilibrium full power is approximately 0.7.

During the subsequent power ascension program, the release rate was above 100 000 uCi/sec. from 9:30 pm to 12:50 am on April 15, 1972 (maximum 126,100 uCi/sec.) The power level was held at 1400 MWt until the release rate was below 100,000 uCi/sec (administrative hold by JCP&L).

C. Action by Licensee:

Mr. McCluskey stated that he had been informed by General Electric that $\bar{E}$ changes after reactor shutdown and would be substantially less than at full power. As a result, the release rate could be substantially higher without exceeding Technical Specification limits. The power ascension program was tailored to keep the release rate below 100,000 uCi/sec. At 11:00 am on April 17, 1972, the release rate was approximately 85,000 uCi/sec. at 1800 MWt.

8304180099 1p.

APR 1 8 1972

J. G. Keppler, Chief, Reactor Testing & Operations Br.
Division of Compliance, HQ

CO INQUIRY REPORT NO. 50-219/72-07
JERSEY CENTRAL POWER & LIGHT COMPANY
OYSTER CREEK - BWR
EQUIPMENT FAILURE - FOUR CONTROL ROD DRIVES SETTLED AT NOTCH "02" ON SCRAM

The subject inquiry report is forwarded for your information.

The cause of the scram appears to have been an operator error, possibly as a result of the failure to use a procedure. The failure of the four control rods to fully insert does not appear to be serious unless it is an indication of degradation of all control rod drives.

The licensee plans to inspect these drives, after they are removed to determine the cause of failure. We consider this course of action to be adequate.

Our inspector will review this matter during the next routine inspection. At present, the licensee plans to report this occurrence along with the high stack activity (Inquiry Report 50-219/72-08) in one letter.

R. T. Carlson
Senior Reactor Inspector

Enclosure:
Subject Inquiry Report

cc: E. G. Case, DRS (3)
R. S. Boyd, DRL (2)
R. C. DeYoung, DRL (2)
D. J. Skovholt, DRL (3)
H. R. Denton, DRL (2)
L. Kornblith, CO
R. H. Engelken, CO
CO Files
DR Central Files

B/385

OFFICE ▶	CO					
SURNAME ▶	Cantrell:smg	Carlson				
DATE ▶	4/19/72					

8304180186 1p.

CO Inquiry Report No. 50-219/72-07

Subject: Jersey Central Power & Light Company

License No.: DPR-16

Facility: Oyster Creek - BWR

Title: Equipment Failure - Four Control Rod Drives Settled at Notch "02" On Scram

Prepared by: F. S. Cantrell, Jr., Reactor Inspector   Date

A. Date and Manner AEC was Informed:

By telephone call from Mr. T. J. McCluskey, Station Superintendent, April 15, 1972 (at home).

B. Description of Particular Event or Circumstance:

A low water level reactor scram occurred on April 13, 1972 at 1:55 pm due to the feedwater (FW) pumps tripping off. A manual turbine trip was initiated, the steam bypass valves opened and eventually the main steam isolation valves closed, thus bottling up the reactor. The minimum water level reached was seven feet seven inches above the active fuel. Reactor pressure initially dropped to 938 psig and increased to a maximum of 1110 psig. The electromatic relief valves (ERV) did not lift, however, the isolation condenser was initiated automatically by reactor pressure remaining above 1070 psig for longer than 15 seconds.

All systems responded properly following the scram except the following:

1. Only four of the five recirculation pumps tripped on the scram. (The trouble was traced to an improperly adjusted contact on the IK77 relay.)

2. Four control rod drives settled at the "02" notch on the scram. Two of these (18-11 and 30-31) were repeats that were previously reported in a letter from JCP&L to DRL dated January 25, 1972. JCP&L plans to replace these drives along with the two additional drives that stopped at "02" (18-27 and 18-35) during the refueling outage that is scheduled to begin on April 22, 1972.

8304180200 2pp.

C. Action by Licensee:

The FW pumps tripped because of misoperation of valves in the radwaste building. A batch of "reprocessed water" was transferred back to the plant via the suction of the condensate pumps. The valves in the transfer line were not closed when the transfer was completed. As a result, air was sucked in by the condensate pumps. The condensate pumps normally deliver water to the FW pumps at 150 psi. Initially two FW pumps tripped on low suction pressure. The operator restarted these two pumps and reactor level was beginning to recover when all three FW pumps tripped. The reactor then scrammed on low level. The FW pumps trip setpoints were checked (47, 49 and 58 psi). The low suction pressure trip is provided to protect the pumps.

3944

APR 13 1972

J. G. Keppler, Chief, Reactor Testing & Operations Br.
Division of Compliance, HQ

CO INQUIRY REPORT NO. 50-219/72-06
JERSEY CENTRAL POWER & LIGHT COMPANY
OYSTER CREEK - BWR
DEPARTURE FROM FSAR/TS - LOSS OF SECONDARY CONTAINMENT CAPABILITY

The subject inquiry report is forwarded for your information and possible action, due to its possible generic applicability.

We believe JCP&L action in this case is acceptable. Mr. McCluskey stated that this work (rack out of breaker and removal of fan motor) was done without a written procedure, and that this subject would be a part of the investigation by the Plant Operations Review Committee and would be included in the report as is appropriate.

We plan to review this event and the related subject of written procedures for safety related maintenance activities during the next inspection of Oyster Creek. We also plan to alert the other BWR facilities within Region I of the possible generic aspects of the problem. We will keep you informed as is appropriate.

R. T. Carlson
Senior Reactor Inspector

Enclosure:
Subject Inquiry Report

cc: E. G. Case, DRS (3)
R. S. Boyd, DRL (2)
R. C. DeYoung, DRL (2)
D. J. Skovholt, DRL (3)
H. R. Denton, DRL (2)
L. Kornblith, CO
R. H. Engelken, CO
Regional Directors, CO
CO Files
DR Central Files

B/386

OFFICE ▶	CO					
SURNAME ▶	Cantrell:smg	Carlson				
DATE ▶	4/13/72					

Form AEC-318 (Rev. 9-53) U.S. GOVERNMENT PRINTING OFFICE

8304180696 2p.

CO Inquiry Report No. 50-219/72-06

Subject: Jer Central Power & Light Company

License No.: DPR-16

Facility: Oyster Creek - BWR

Title: Departure from FSAR/TS - Loss of Secondary Containment Capability

Prepared by: ______________________ ____________
F. S. Cantrell, Jr., Reactor Inspector Date

A. Date & Manner AEC was Informed:

By telephone call from Mr. T. J. McCluskey, Station Superintendent, on April 11, 1972.

B. Description of Particular Event or Circumstance:

The 1-13 supply fan for reactor building ventilation was removed from service and its power breaker racked out in order to remove the motor for maintenance on April 10, 1972. During a surveillance test of the high radiation sensors on the operating floor of the reactor building on April 11, 1972, the standby gas treatment system started as required, however, the dampers associated with the supply fans failed to close to complete the isolation of the reactor building. Technical Specifications do not restrict the removal of one fan from service, however, the reactor building isolation circuit and associated equipment must be operable.

C. Action by Licensee:

An investigation showed that when one fan breaker is racked out, as is required to replace the motor, the isolation circuit is rendered inoperable. (If a fan trips off, the isolation circuit remains operable.) The motor leads were lifted and the breaker was racked in, thus, making the isolation circuit and dampers operable.

General Electric had submitted a design change for the circuit to Jersey Central prior to this event; however, the proposed change was still under review by the licensee at the time of this event.

Mr. McCluskey stated that the TS would be reviewed to determine if and how the fan can be returned to service or if the fan can remain out of service until the refueling outage, scheduled to begin April 22, 1972.

Mr. McCluskey stated that a written report would be made to DRL within ten days as required by TS.

8304180704 1p.