U. S. ATOMIC ENERGY COMMISSION
DIVISION OF COMPLIANCE
REGION I

CO Inspection Report No. 50-219/72-02

Subject: Jersey Central Power & Light Company

Oyster Creek

License No. DPR-16

Location: Forked River, New Jersey

Priority

Category C

Type of Licensee: BWR (1930 MWt)

Type of Inspection: Routine

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

Dates of Previous Inspection: Routine July 2, 1971
Special February 7, 1972

Principal Inspector: F. S. Cantrell, Reactor Inspector — Date

Accompanying Inspectors: None — Date

Date

Other Accompanying Personnel: None — Date

Reviewed By: R. T. Carlson, Senior Reactor Inspector — 5/16/72 Date

Proprietary Information: ne

Section I

Enforcement Action:

A. Technical Specification 3.4.C.5 specifies in part, "During the period when one diesel is inoperable, the containment spray loop....connected to the operable diesel shall have no inoperable components".

On January 14, 1972, while the No. 1 containment spray system (which is connected to the No. 1 emergency diesel generator) was inoperable in the course of a scheduled surveillance test, the No. 2 emergency diesel generator was made inoperable to permit adding oil. This condition existed for 45 minutes. (Paragraph 7)

B. Technical Specification 6.6.B requires in part, "The events listed below require reports within 24 hours by telephone or telegraph to Region I Compliance office followed by a written report within 10 days to the Director, Division of Reactor Licensing....2.Any abnormal occurrence as specified in Section 1.15..". Section 1.15.B defines an abnormal occurrence as "Violates a limiting condition for operation as established in Section 3 of the Technical Specifications, or..".

Contrary to the above requirement, the written report of the violation was not submitted until February 22, 1972. (Paragraph 7 and Management Interview)

Licensee Action on Previously Identified Enforcement Matters:

As a follow up to the April 1971 inspection, a formal enforcement letter was sent to the licensee from CO:HQ on August 25, 1971 identifying three items of noncompliance with Regulatory requirements pertaining to the release and storage of liquid radioactive waste and two other issues involving variances in the operation of the facility from information presented in the FD&SAR.

In a letter dated September 16, 1971, JCP&L replied to the enforcement action as follows:

A. Corrections were made in the sampling and analytical techniques (discussed in CO Report No. 219/71-2). Corrected figures for total radioactivity released were supplied to the Division of Reactor Licensing in a letter dated September 22, 1971 that accompanied Semi-Annual Report No. 4. Revised pages for Semi-Annual Reports Nos. 1, 2, and 3 were included. This item is considered resolved.

B. The source of the excessive background radiation in the vicinity of the liquid effluent monitor was removed. This item is considered resolved.

C. Jersey Central did not agree with the requirement to report any substantial variance disclosed by operation of the facility from performance specifications contained in the Facility Description and Safety Analysis Report (FD&SAR) or the Technical Specifications (TS). This item was resolved in a letter to Jersey Central from L. D. Low, Director, Division of Compliance, dated December 30, 1971, re-affirming the requirement to report any substantial variances.

As a result of the June 23 - 25, 1971 and July 2, 1971 inspection, two items of noncompliance with Regulatory requirements were identified in a letter from J. P. O'Reilly, Director, Region I, to JCP&L on September 14, 1971.

In a letter dated October 1, 1971, JCP&L replied to the enforcement action as follows:

A. The incorrect trip point setting for the radiation monitor in the main steam line tunnel was detected by a General Office Review Board (GORB) audit and was corrected prior to the inspection.

   Item resolved at time of subject inspection, as noted in September 14, 1971 letter.

B. New administrative procedures were reported to have been instituted which require the General Public Utilities Safety and Licensing Group to review all GORB audit reports for licensing violations and to report the results to the Chariman, GORB. The Chairman reports separate violations to the President, JCP&L as necessary.

   The JCP&L letter further stated, "With respect to Item No. 2 of your letter, an investigation will be conducted and the results reviewed at the next GORB meeting....".

   Contrary to the above commitment, the minutes of the next GORB meeting, which was on November 23, 1971, did not show that the investigation had been conducted or that the results had been reviewed. Records did not indicate any other meeting had been held subsequent to October 1, 1971.

As a follow up to the November 19, 1971 inspection, one item of noncompliance with Regulatory requirements was identified in a letter to JCP&L from J. P. O'Reilly, Director, Region I on December 23, 1971. No reply was requested since corrective action was initiated and the violation was reported in a letter from JCP&L dated December 14, 1971.

Unresolved Items: None

Status of Previously Reported Unresolved Items:

A. QA records for the installation of relief valve No. NR108E are at the site. No deficiencies were noted. This item is considered resolved. (Paragraph 5.i)

B. QA records are at the site for: 1. the six new 10 inch swing check valves in the core spray system (Paragraph 5.h); 2. two new Powell valves in the poison system (Paragraph 5.k); 3. eight isolation condensers drain valves (two each drain line) (Paragraph 5.g). These valves were installed during the September - November 1971 outage. This item is considered resolved.

C. A new isokinetic probe was installed in the plant stack. This area was not inspected.

D. A jockey pump (to keep the piping system filled) has been installed on both core spray loops. A preliminary report of the investigation of the core spray water hammer was submitted to the Division of Reactor Licensing on June 25, 1971. (Paragraph 8)

E. Reactor vessel level instrumentation - The "A" GE/MAC level indicator does not agree with the "B" GE/MAC or the Yarway level indicators (1.3 feet lower). This problem is still unresolved. (Paragraph 9)

F. The basis for setting the 45% bypass device for turbine scrams was provided in Amendment No. 65 (Application to Increase Power Level), approved on November 5, 1971. This item is considered resolved.

G. The protective devices for the emergency diesel generator (EDG) when in the fast start mode were functionally tested during the annual inspection. These devices for the No. 1 EDG were calibrated in February 1972. No. 2 is scheduled in April 1972. (Paragraph 12)

Unusual Occurrences:

A. The generator load rejection and turbine trip anticipatory scram bypass switch failed due to a packing leak on the root valve on August 2, 1971 (Letter, JCP&L to DRL, dated September 9, 1971).

B. The standby gas treatment system train No. 1, minimum flow valve failed to open due to a solenoid valve failure on July 6, 1971 (Letter, JCP&L to DRL, dated September 9, 1971).

C. One of the four scram dump volume level switches failed due to binding during a surveillance test on August 17, 1971 (Letter, JCP&L to DRL, dated September 9, 1971).

D. While performing a surveillance test on the B isolation condenser line break isolation sensors, the time delay feature of both isolation relays was found inoperable thus negating the automatic isolation function in the event of a line break (Letter, JCP&L to DRL, dated September 30, 1971). (Paragraph 10)

E. As a result of an accident involving one of the mobile nitrogen evaporators used to inert containment and severe weather conditions where the backup nitrogen evaporator was located, equipment was not available to inert the containment within 24 hours after the reactor was placed in the run mode on November 23, 1971. A temporary Technical Specification change was approved by the Division of Reactor Licensing to permit operation at up to 50% of full power with the O_2 content greater than 5% for an additional 24 hours. (Letter, JCP&L to DRL, dated November 22, 1971 and letter, DRL to JCP&L, dated November 22, 1971) (Paragraph 11)

F. Following the annual inspection on September 9, 1971, the generator breaker was inadertantly closed on a live bus with the diesel engine at a standstill. (Letter, JCP&L to DRL, dated December 13, 1971)

G. The vent line from the isolation condenser to one of the main steam lines broke, downstream of the isolation valves in both the vent line and the main steam line. The reactor was shutdown and the line was repaired by welding. (Letter, JCP&L to DRL, dated January 12, 1972)

H. During a closure test of the main steam isolation valves on September 18, 1971, one of the inside valves closed faster than desired (3.2 seconds). An oil leak in the hydraulic dash pot adjustment leg was determined to be the cause. During a subsequent test, the leakage through the valve was greater than TS limits. The valve stem was straightened and the main and pilot valve seats were lapped (Letter, JCP&L to DRL, date d December 13, 1971).

I. During a routine surveillance test on December 28, 1971, the No. 1 emergency diesel generator (EDG) tripped off when it ran out of fuel in the "day tank". The starting switch on both transfer pumps was dirty and corroded. (Letter, JCP&L to DRL, dated January 5, 1972) (Paragraph 12)

J. The radionuclide inventory in the outside tank farm exceeded 0.7 Ci on several occasion (TS limit 0.7 Ci) and was recycled until compliance with the TS limit was achieved as required by the TS. Due to problems with the waste concentrator plugging, rapid depletion of the rad waste demineralizer and the large inventory remaining from the September - November 1971 outage, the inventory of the tank farm exceeded 0.7 Ci from November 29 to December 17, 1971. (Letter, JCP&L to DRL, dated December 22, 1971) (Paragraph 13)

K. While the No. 1 containment spray system was out of service for surveillance testing, the No. 2 emergency diesel generator that powers the No. 2 containment spray system was removed from service for servicing. (Letter, JCP&L to DRL, dated February 22, 1972) (Paragraph 7)

L. While testing the "normal emergency power DC interlock failure" alarm on January 22, 1972, DC power was transferred from 125V DC Bus A to 125V DC Bus B. The momentary loss of power caused a trip of three of the five recirculation pumps and one feedwater pump. Plant power dropped from 642 MWe to 316 MWe and leveled at 336 MWe. (Letter, JCP&L to DRL, dated February 22, 1972)

M. While removing the "A" battery motor generator set from service for maintenance, the main breaker was opened before the static charger was closed in on the bus. The loss of DC voltage on A bus caused three of the five recirculation pumps, and one feedwater pump to trip. Reactor power dropped from 640 MWe to 346 MWe and leveled off at 400 MWe. (Letter, JCP&L to DRL, dated February 23, 1972)

N. During a check out of repairs to the current transformers that supply overload protection to the 1A auxiliary transformer, power to the 1C emergency bus tripped off due to an error in setting up the test. The operator attempted to restore power by reclosing the normal supply at the same time the emergency diesel generator (EDG) phased on line. The EDG tripped on reverse current and the normal supply tripped because the fault that was set up in the transformer check out was still in the system. (Paragraph 14)

O. One of the four scram dump volume level switches failed during a surveillance check on March 1, 1972 due to dirt and water in the switch assembly that prevented full travel of the switch arm. (Letter, JCP&L to DRL, dated March 10, 1972)

<u>Persons Contacted:</u>

T. J. McCluskey, Station Superintendent
J. T. Carroll, Operations Supervisor
R. M. McKeon, Shift Foreman
R. VanBrakle, Control Room Operator
J. Glendenning, Control Room Operator
D. A. Ross, Technical Supervisor
E. I. Riggle, Maintenance Supervisor
J. L. Sullivan, Technical Engineer
D. Pelrine, Chemical Supervisor
D. E. Kaulback, Radiation Protection Supervisor
K. O. Fickeissen, Assistant Technical Engineer

Management Interview:

The inspector conducted an exit interview on March 1, 1972 with Messrs. McCluskey, Ross, Carroll and Riggle.

A. Comments on Discussion with Operators

Mr. McCluskey stated that he was not aware of any problems from the inspector's discussion with two operators on the loss of the 1C bus that occurred on December 22, 1971. The two operators were on duty at the time. Mr. McCluskey asked the inspector if he had any problem. The inspector stated that the two men had been very cooperative and to the best of the inspector's understanding, verified the information in the OC internal report to the PORC.

B. Violation of LCO No. 1, Containment Spray System, and No. 2, EDG Out of Service Simultaneously (1/14/72)
(Letter, JCP&L to DRL, dated February 22, 1972)

According to plant records, the violation was discovered on January 15, 1972, but was not reported to Region I until January 26, 1972, and as of the date the inspector left his office on February 22, 1972, the written report had not been received by Region I.

Mr. McCluskey stated that normally when testing the core spray system, the core spray system is not inoperable - an initiation signal will override the test signal; however, for this particular test, it was necessary to close and de-energize a valve that made the system inoperable. He stated that even though he was informed on January 15, 1972 that both systems were off line at the same time, he did not realize that a LCO was violated until a full investigation was completed on January 25, 1972 and at that time he reported the violation to Compliance. He stated he could not explain the reason for the delay in the written report other then the time necessary to get a report through the necessary channels. Mr. Ross obtained a copy of the written report (dated February 22, 1972) that was sent to DRL. After some discussion, the inspector stated he accepted the explanation for the delay in the verbal report to Compliance; however, the failure to submit a written report of the violation of an LCO within 10 days after it was discovered was considered in noncompliance with the Technical Specifications. (Paragraph 7)

C. Exceeding TC Limits (0.7 Ci) for Tank Farm Inventory

The inspector discussed the present controls which require the plant chemist to notify the shift foreman when the results of his analysis show that the 0.7 Ci limit is exceeded.

Mr. Carroll stated that additional information would be provided to the shift foreman in order to reduce the number of violations. (Mr. McCluskey subsequently informed the inspector by telephone that the results of the Monday - Wednesday - Friday inventories are reported to the shift foreman to help him plan his reprocessing schedule.) (Paragraph 13)

D. Loss of 4160 Volt Emergency Bus 1C

The inspector stated that paragraph 3.7.A.1.a of the Technical Specifications (TS) requires the 1C breaker to be energized and paragraph 3.7.B requires placing the reactor in a cold shutdown condition if the requirements of paragraph 3.7.A are not met. Since a violation of a limiting condition for operation (LCO) is defined as an abnormal occurrence, the event requires a 24 hour telephone or telegraph report to Compliance, and a 10 day written report to DRL.

Mr. McCluskey stated that the event had been investigated by the Plant Operations Review Committee (PORC) and the committee had concluded that the event was not a violation of an LCO. The basis for the TS states that the objective is to assure an adequate supply of power with at least one active and one standby source of power for operation of equipment that is required for a safe plant shutdown and to operate the required engineered safety equipment. In light of the safety analysis that one standby source of power (emergency diesel generator) will supply adequate power to place the plant in a safe condition, the PORC interpreted that Specification 3.7.B only applied to the availability of outside power supplies as specified in paragraphs 3.7.A.2 and 3, and station batteries as specified in paragraph 3.7.A.4.

The inspector stated that he felt the words of the TS were specific; however, he would review this matter with his supervision to determine the intent of the specification and how it should be applied. Assuming the specific requirements apply, the following comments are applicable:

1. Emergency procedure No. 502, which covers the loss of power, was revised as a result of the loss of bus 1C: however, neither the old nor the revised procedure recognized the TS requirement to place the plant in a cold shutdown condition when a LCO is exceeded.

2. Written procedures were not used to check out the modifications to the auxiliary transformer.

3. There was no record that the modifications to the auxiliary transformer were approved by the PORC or that the change did not involve an unreviewed safety question. Prior to this event, we probably would have agreed that this type change could have been made without a written safety evaluation as required by 10 CFR 50.59; however,

The inspector stated that PORC meeting minutes did not indicate that the problem of the difference in the reactor level as sensed by the two GE/MAC level indicators had been discussed.

Mr. McCluskey stated that this was apparently an oversight in the minutes since the problem had been assigned to one of Mr. Ross's engineers as an "action item" and that he is required to report his findings to the PORC. Mr. Ross stated the problem was still active. This explanation was accepted. (Paragraph 15)

F. Information Recorded in the Shift Foreman's Log and the Control Room Log

The inspector stated that he did not feel that sufficient information was being recorded in these two logs to serve as a record of significant events. Neither log indicated the reason for the January 27, 1972 shutdown (the unexplained leakage rate increased to 4 gpm) and neither log showed that the stack release rate increased to 142,000 uCi/sec for a short period on January 23, 1972. (The release rate had been about 55,000 uCi/second.)

Mr. Carroll agreed more information should be entered in these two log books and that he would initiate the necessary action. (Paragraph 16)

G. Additional Areas Reviewed to Determine Compliance with Regulations

The inspector stated that he had made a specific review of logs, charts and records to verify: 1. that reactor power had not been increased above 50% until the oxygen in the drywell was less than 5% on November 23, 1972 (Paragraph 11), and 2. that an additional operator was present to act in place of the inoperable rod worth minimizer during the reactor startup on November 11, 1971. The review did not show any items of noncompliance in these two areas.

in retrospect the events that occurred demonstrated that modifications in the power supply area can directly affect plant safety equipment.

The inspector asked if Jersey Central would operate with only the EDG supplying either of the emergency buses (Bus 1C or 1D). After a brief discussion Mr. McCluskey stated that the plant would not be operated with only the EDG available to supply power to its associated emergency bus. He pointed out, however, that the TS permit a startup transformer and an EDG to be out of service simultaneously for seven days. The inspector stated he intended to pursue this subject to determine the intent of the specification.

After reviewing the above event, the inspector called Mr. McCluskey on March 6, 1972 and informed him that TS 3.7.A and B must be interpreted as written and that a written report of the event must be made to DRL. The review indicated that there is some inconsistency between the basis for the TS and the words in TS 3.7.A and B. He was told that Compliance was not implying that with the 1C bus de-energized for five minutes he should have shut down the reactor; however, if the bus had been lost for five hours, the reactor should have been in the process of shutting down. The inspector stated that as long as the present specifications were applicable, Jersey Central will be expected to follow the specification as written; however, if the written report is made within 10 days of the present date, there would be no citation for failure to report this event. (A written report was submitted to DRL on March 10, 1972.)

The inspector pointed out that if there is any doubt as to whether an event is reportable that if the Regional Compliance office is notified within 24 hours, the licensee automatically has at least 10 days by his TS to make up his mind as to whether a written report to DRL is required. (Paragraph 14)

E. Plant Operations Review Committee

The inspector stated he was concerned with the poor attendance of the representatives of the General Office Review Board (GORB) at the PORC meetings. The records indicated that the GORB members only attended two of the monthly meetings during the period July - December 1971.

Mr. McCluskey stated that the inspector's comments would be relayed to the appropriate people. He stated that the investigation of the loss of emergency bus 1C had been delayed a few days to permit the GORB members to be present at the investigation (PORC meeting on January 11, 1972).

Section II

Additional Subjects Inspected, Not Identified in Section I, Where No Deficiencies or Unresolved Items were Found

1. General

Since the July 2, 1971 inspection, the reactor has experienced one scram (complete loss of instrument air on November 16, 1971), two unplanned shutdowns (broken isolation condenser vent line on December 11, 1971 and an increase in unidentified leakage in the drywell on January 28, 1972) and one scheduled shutdown (September 18 through November 11, 1971). During the September - November outage, the licensee completed the following tasks: removed all poison curtains, performed incore sipping of 548 of 560 fuel bundles, inspected 33 suspected leaking fuel bundles, reconstituted 20 fuel bundles, installed 24 new fuel bundles, replaced 7 local power range monitor strings, replaced 1 control blade, replaced 3 control rod drives, inspected the high pressure turbine and the "A" low pressure turbine, replaced 6 of the 9 main steam bypass valve seats, and installed a fifth main steam relief valve in the drywell.

Amendment No. 3 to the facility license was approved on November 5, 1971 permitting operation at 1930 MWt. During the power ascension program, the main steam bypass valves started opening at 1850 MWt indicating that the steam produced at that level was the maximum that could pass the control valves without completion of modifications to the second stage reheaters. This work was started during the September - November outage. Completion is now scheduled for the April 1972 refueling outage. All of the testing that was scheduled for 1930 MWt was then performed at approximately 1830 MWt except for the five recirculation pump trip test and the turbine trip test. The five recirculation pump trip test was performed on January 27, 1972 just prior to the shutdown for high drywell leakage on January 28, 1972. The turbine trip test is scheduled to be performed just prior to the shutdown for refueling (April 1972).

Following the January 28, 1972 shutdown, the effluent cooling water dropped to approximately 35° F, a decrease of 25° F during the shutdown. A sudden change in the weather prior to the shutdown dropped the cooling water temperature approximately 10° F during the two days just prior to the shutdown. Newspapers in the area reported thousands of dead fish as a result of the plant shutdown (CO Report No. 50-219/72-01).

Records show that the stack release rate, which was approximately 50,000 uCi/second prior to the September 1971 shutdown, decreased during steady state operation after the September - November 1971 outage to approximately 40,000 uCi/second, but has since increased to approximately 55,000 uCi/second with peaks as high as 142,000 uCi/second after moving control rods (for flux shaping and reactivity control).

2. Administration and Organization

   a. Personnel changes.

   b. Plant Operations Review Committee (PORC) meeting minutes July 29, 1971 through February 10, 1972. (except as noted in paragraph 15)

   c. General Office Review Board (GORB) meeting minutes and GORB audit reports August 17, 1971 through February 10, 1972.

3. Operations

   a. Abnormal Occurrence Records - August 1971 through February 1972.

   b. Plans to examine main steam line flow restrictor sensing lines.

   c. Program for testing torus to drywell vacuum breaker valves.

   d. Records of reactor vessel thermal cycles.

4. Procedures

   a. Small Leak Detection (No. 515)

   b. Emergency Diesel Generator Monthly Inspection (No. 726.2)

   c. Diesel Generator 20% Plus Load Test

5. Maintenance

   a. Modifications to electromatic relief valves.

   b. Plans to replace two control rod drives that failed to insert.

   c. Plans to inspect the baffles in the torus during the April 1972 refueling outage.

   d. Installation of a "jockey pump" for the No. 1 core spray loop.

   e. Depth of lifting holes in the standby liquid control pump.

   f. Off gas isolation circuit.

   g. Quality control records for installation of isolation condenser drain valves.

h. Quality control records for installation of core spray system check valves.

i. Quality control records for installation of the fifth relief valve in the main steam system.

j. Repairs to the isolation condenser vent line.

k. Quality control records for the installation of two valves in the poison system.

l. Modification to the electromatic relief valves.

6. Surveillance Testing

a. Main steam isolation valve closure time test.

b. Records of reactor pressure vessel thermal cycles.

c. Testing of torus to drywell check valve.

d. Emergency service water pumps.

Details of Subjects Disoussed in Section I

7. Violation of LCO No. 1 Containment Spray System and No. 2, EDG Out of Service Simultaneously
(Letter, JCP&L to DRL, dated February 22, 1972)

The control room log shows that the No. 1 containment spray system was taken out of service at 8:10 a.m. on January 14, 1972, the No. 2 EDG mode switch was placed in the "off" position at 8:20 a.m., and the No. 1 containment spray system was returned to service at 9:05 a.m. on January 14, 1972.

Technical Specification paragraph 3.4.C.5 specifies in part, "During the period when one diesel is inoperable, the containment spray loop....connected to the operable diesel shall have no inoperable components." The No. 1 containment spray loop is powered by the No. 1 EDG.

This violation was reported to Region I by telephone on January 26, 1972. A written report was made to DRL on February 22, 1972.

8. Core Spray Jockey Pump

A field inspection showed that JCP&L has completed the installation of a jockey pump (to keep the system filled) on both core spray loops. (Reference CO Report 219/71-2, paragraph 11). This item is considered complete.

9. Reactor Vessel Level Instrumentation
(CO Report 219/71-2, paragraph 20)

The A GE/MAC level indicator indicates that water level is approximately 1.3 feet lower than either the B GE/MAC or the Yarway indicators. This was discussed with Mr. Riggle. The two systems were recalibrated completely during the September - November 1971 outage, the difference remains. The venting of the condensing pots was discussed - the pots are not vented. Mr. Riggle said the problem had been reviewed with the PORC; however, meeting minutes did not confirm this statement. During the exit interview, other members were aware of the problem and agreed that the PORC minutes should have included a discussion of the work on the level indicators, since one of Mr. Ross's engineers had been assigned to investigate the problem and report back to the PORC. The issue is still unresolved.

10. Isolation Condenser Relay Failure
(Letter, JCP&L to DRL, dated September 30, 1971)

The relays that failed were initially described as GE CR120 type relays with a time delay addition. This information was provided in a telephone report to Compliance on September 9, 1971 (Inquiry Report 219/71-05). Jersey Central was asked by telephone on February 9, 1972 to provide more specific information as to the type of relay replaced. On February 14, 1972, the inspector was told that the relay specifically was a CR122A-09041AA (time delay) relay.

In response to a question as to how the wrong information was provided, Mr. Riggle stated that the CR120 relay was the basic relay and the modification made for Oyster Creek made the relay a CR122A-09041AA. The initial look had just shown the CR120 designation. GE literature describing CR120 and CR122 relay was reviewed; however, the literature did not include a description of the above numbered relay.

Mr. Riggle stated that numerous other CR120 type relays are used at Oyster Creek but no CR120's are used directly in the reactor protection system. CR120's as used in allied systems such as the reactor manual scram indicating lights circuit, scram relay resets circuits, scram discharge volume high level bypass circuit, and condenser low vacuum bypass circuit.

11. O_2 Concentration in Drywell

A temporary TS change dated November 23, 1971 permitted operation at 50% of licensed power with O_2 concentration in the drywell - torus greater than 5% for 48 hours until 2:11 a.m. on November 24, 1971. (Weather conditions and a wreck prevented getting "N_2 pumps" to the site within the prescribed 24 hours.)

The control room log showed that the oxygen content of the drywell decreased to $<5\%$ at 10:18 a.m. on November 23, 1971 and that reactor power was at 935 MWt (approximately 48%) until 10:18 a.m. At that time, a power increase was initiated using recirculation flow controls. Recorder charts and data sheets for November 23, 1971 verified the entry in the control room log book.

Plans for the installation of permanent equipment at the site for inerting the drywell and torus were discussed. Mr. Riggle stated that all of the equipment needed was at the site and was in the process of being installed.

12. Emergency Diesel Generator out of Fuel
(Letter, JCP&L to DRL, dated January 5, 1972)

The subject failure was discussed with Mr. McKeon and Mr. Carroll, and the equipment involved was inspected. A review of the procedure for the monthly inspection verified that the procedure had been modified to check the fuel oil transfer pumps and to verify that the trouble alarm (for the EDG) operated in the control room as was reported in the referenced letter.

In response to a question about protective devices on the EDG and the testing calibration of same, Mr. McCluskey provided the following in a subsequent telephone call:

Shutdown devices that are bypassed in the fast start mode:

a. Engine temperature high switch
b. Main bearing oil pressure high switch
c. Main bearing oil pressure low switch
d. Overspeed trip limit switch (back up)
e. Lube oil temperature low switch
f. Low water pressure switch

Devices that unload diesel, open generator breaker and throttle diesel back to "idle" speed:

g. 1. No starter pinion engagement
   2. No engine start
   3. No voltage buildup
   4. Generator overvoltage
   5. No generator circuit breaker closed

Devices that open generator breaker:

h. Generator breaker relay

   1. Undervoltage relay
   2. Leading vars relay

3. Reverse power relay
4. Phase differential relay
5. Engine overspeed relay

Devices that stop the diesel:

i. Overspeed trip on the diesel

The devices in g, h and i were fast start tested during the annual inspection in September 1971. The relays for No. 1 EDG were calibrated during February 1972 by the JCP&L Relay Department. The relays in No. 2 EDG are scheduled to be calibrated during April 1972.

13. Rad Waste Tank Farm Inventory
(Letter, Jersey Central to DRL, dated December 22, 1971)

Technical Specifications limit the inventory to 0.7 Ci (paragraph 3.6.C). If the limit is exceeded, TS require reprocessing until the inventory is less than 0.7 Ci.

This limit is exceeded frequently according to the records; however, normally the condition is corrected within a few hours by recycling. As this is an LCO, a written report is required to be sent to DRL. Following the September - November 1971 outage, larger than normal amounts of liquid rad waste were on hand. When the inventory was calculated beginning November 14 (calculated Monday-Wednesday-Friday), the limit was exceeded. This continued through December 17, 1971. On November 29, 1971, the contents of the tank farm could not be recycled due to the plugging of the tubes of the waste concentrator and the rapid depletion of the rad waste demineralizer. Three tank truckloads (3700 gallons each) were transferred to a licensed carrier for disposal by Nuclear Engineering Company in Kentucky, releases were made to the environment in accordance with the TS, the tubes of the waste concentrator were drilled out, and the demineralizer was regenerated. The inventory was reduced to less than 0.7 Ci on December 20, 1971.

Plant operation was allowed to continue with the verbal concurrence of CO and DRL for the following reasons:

a. Jersey Central indicated they were doing all possible to get the waste concentrator and the demineralizer back in service.

b. The inventory was being reduced by off plant shipments.

c. A reactor shutdown would not have stopped additions to the rad waste system.

d. The condition could have been corrected by releases to the environment at the maximum rate permitted by the TS.

It was concluded that the proper action in light of the requirement to keep releases "as low as practical" was to allow the inventory in the tank farm to remain above 0.7 Ci while the waste concentrator and demineralizer were repaired or regenerated.

Records show that an inventory is calculated Monday, Wednesday and Friday (TS 4.6.D requires analysis every 72 hours). If the results indicate the limit is exceeded, a report is sent to the shift foreman with recommendation as to the priority of reprocessing the various tanks. In response to questions by the inspector, Mr. Carroll stated that additional information would be provided to the shift foreman to enable them to reduce the number of violations. (Previously the shift foremen were not advised of the inventory status until the limit was exceeded.)

14. Loss of 4160 Volt Emergency Bus 1C
(Letter, JCP&L to DRL, dated March 10, 1972)

The reactor control room log showed that the 1C breaker opened at 11:45 a.m., December 22, and initiated a "1/2 scram" on the No. 1 reactor protection system. The log showed that the electrical system was returned to normal at 12:10 p.m. the same day. The two operators that were on duty when the 1C breaker opened were interviewed as were Mr. Carroll and Mr. Riggle.

While checking circuit continuity for a spare set of current transformers to be used as replacements for the burned out units associated with the auxiliary transformer, the proper fuses were not removed to isolate the circuit. When a voltage signal was applied to the circuit, protective relay actions occurred that tripped the 1C breaker. Emergency diesel generator No. 1 was initiated in the "fast start" mode and had just re-energized the 1C bus when a reclosure of the 1C breaker was attempted. The 1C breaker was only closed for a few cycles since the false signal had not been removed; however, during that time, the voltage and frequency mismatch activated the EDG reverse power relay, tripping the unit. As a result, the 1C bus was de-energized for approximately five minutes while the false signal was removed and normal power was being restored to the bus. The remainder of the twenty-five minutes was consumed in restarting equipment.

According to Mr. Riggle, a written procedure was not prepared for check out of the current transformer since written procedures are not specifically required in the area of power distribution. The mechanics involved in the check out pulled the wrong fuses and failed to isolate the circuit.

In response to a specific question, the persons interviewed stated that no consideration was given to shutting down the reactor during the period the 1C bus was de-energized. Both emergency buses are required to be energized in TS 3.7.1 (limiting condition for operation - LCO).

The loss of the 1C bus was investigated by the Plant Operations Review Committee (PORC) on January 13, 1972. A report was prepared and the PORC concluded that the event did not reduce the availability of power as provided for in the Technical Specifications.

As a result of this event, the emergency procedure (No. 502) for Loss of Power was revised. Neither the original procedure or the revised procedure acknowledge the loss of an emergency bus as a violation of a LCO. This subject was discussed during the exit interview.

15. Plant Operations Review Committee (PORC) Meetings

Meeting minutes show that 18 meetings were held during the period July 1971 to February 1972, which adequately met the TS requirement (paragraph 6.1.C). Two members of the General Office Review Board (GORB) are appointed by the Chairman of the GORB as members of the PORC. The PORC meeting minutes showed that the GORB members were present at two of the meetings during the period July 1971 to December 1971. (GORB was represented at two of the three meetings in 1972.) The TS do not specifically require the GORB members to be present.

The PORC meeting minutes did not show that the differences in reactor level instrumentation had been reviewed by the PORC. The members of the staff stated during the exit interview that this was an oversight in keeping the minutes. Mr. Ross stated that one of his engineers had been assigned to this problem by the PORC.

16. Shift Foreman's Log and Control Room Log

Portions of these two logs were read for the period June 25, 1971 through February 27, 1972. The information provided was sketchy. Two noteworthy events were not recorded: a. the reason for the planned shutdown on January 27, 1972 (an increase to 4 gpm in the unexplained leak rate), and b. an approximate 150% increase in stack release rate to 142,000 uCi/sec on January 23, 1972 (previous release rate was 50 - 60,000 uCi/sec).

This item was discussed with Mr. Carroll and he agreed that more information should be recorded in the two logs. He stated that he had recently discussed this subject with the shift foremen and believed the logs now reflect better coverage of the plant operation, and he would continue to push for improvement in this area.

17. Failure of Two Control Rods to Fully Insert (18-11 and 30-31)
(Letter, JCP&L to DRL, dated January 25, 1972)

Mr. McCluskey re-affirmed plans to replace the drives for these two rods during the outage scheduled to start approximately April 22, 1972. Rebuilt drives will be installed. The two drives will be dismantled and inspected after the outage.

18. Stack Release Rates

Release records were reviewed and the stack monitor chart for January 1972 was unrolled and inspected. The records show a maximum stack release rate of 142,800 uCi/second for about one hour on January 23, 1972. This rate occurred following the return to power from a runback on January 22, 1972 (loss of three recirculation pumps and one feedwater pump). During routine level operation, the release rate has been approximately 55,000 uCi/second.

The release rate as shown on the stack activity recorder is recorded hourly in counts per second (cps) along with wind speed and direction. The recorder is set to alarm when the cps increase to the equivalen of 50% of the maximum allowable release rate. An off gas sample is analyzed at least weekly to calibrate the stack monitor. This calibration is used to determine the stack release rate in uCi/second. The rates shown on the attached table are typical for January - February 1972.

The inspector questioned the lack of any note in the Control Room Log or the Shift Foreman's Log about the increase in release rate to 142,800 uCi/second. This type increase is normal following a return to power according to Mr. Carroll, and this was probably why a special note was not made in the log books.

Stack Release Rates

(Typical Rates for January - February 1972)

Date	Release Rate (uCi/cc)
1/5	56,700
1/12	40,400
1/20	65,100
1/21	62,500
	← Lost 3 "Recirc" and 1 feedwater pumps
1/22	136,000
1/23	142,800
2/2	43,900
2/8	55,000
2/16	55,400
2/23	55,400

May 9, 1972

J. G. Keppler, Chief, Reactor Testing & Operations Br.
Regulatory Operations, Headquarters

RO INQUIRY REPORT NO. 50-219/72-12
JERSEY CENTRAL POWER & LIGHT COMPANY
OYSTER CREEK 1 - BWR
HUMAN ERROR - FUEL BUNDLE FOUND WITH A 90° MISORIENTATION

The subject inquiry report is forwarded for your information.

Our preliminary review of this matter, which included the licensee's FSAR, disclosed that operation with fuel assembly misorientations of 90° are less severe in consequences than the analyzed case involving 180° misorientations. We believe the licensee's intended course of action is appropriate.

We will continue to follow this matter and keep you informed as appropriate. In addition, our inspector will review this matter, as well as any noncompliance aspects, during the next site inspection. The licensee will submit a written report (10-day) to RL.

R. T. Carlson
Senior Reactor Inspector

Enclosure:
Subject Inquiry Report

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

B/379

~~8303310522~~ 1p.

OFFICE ▶	RO					
SURNAME ▶	SPESSARD:maz	CARLSON				
DATE ▶	5/9/72					

RO Inquiry Report No. 50-219/72-12

Subject: Jersey Central Power & Light Company

License No: DPR-16

Facility: Oyster Creek 1 - BWR

Title: Human Error - Fuel Bundle Found with a 90° Misorientation

Prepared by: ______________________ ______________
T. Young, Jr., Reactor Inspector Date

A. Date & Manner AEC was Informed:

By telephone call from Mr. Tom McCluskey, Station Superintendent, on May 8, 1972.

B. Description of Particular Event or Circumstance:

During the current outage while performing fuel sipping operations, one fuel bundle (25-08) was found in a position of 90° misorientation. This bundle was last handled on October 31, 1971. OC-1 procedures call for a verification of fuel bundle orientation before vessel head closure. This check was made by T.V., recorded on film, and the film was reviewed. A subsequent review of the film on May 8, 1972, clearly showed the bundle with the 90° misorientation.

C. Action by Licensee:

1. The licensee is making an analysis of the consequences of operating with the fuel bundle in the 90° misoriented position.

2. A visual inspection of the bundle will be conducted.

3. A 10-day written report will be made to RL.

~~8303310526~~ 1p.

MAY 8 1972

J. G. Keppler, Chief, Reactor Testing & Operations Br.
Regulatory Operations, HQ

RO INQUIRY REPORT NO. 50-219/72-11
JERSEY CENTRAL POWER & LIGHT COMPANY
OYSTER CREEK - BWR
EQUIPMENT FAILURE - LOOSE BAFFLES IN TORUS

The subject inquiry report is forwarded for your information.

This inspection was made in response to a written request from RL, and the inspection results will be reported in writing to RL as requested in their letter.

We plan to review this matter during the next site inspection.

R. T. Carlson
Senior Reactor Inspector

Enclosure:
Subject Inquiry Report

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

3/3/80

~~8303310558~~ 1p.

OFFICE ▶	RO					
SURNAME ▶	Spessard:smg	Carlson				
DATE ▶	5/8/72					

RO Inquiry Report No. 50-219/72-11

Subject: Jersey Central Power & Light Company

License No.: DPR-16

Facility: Oyster Creek - BWR

Title: Equipment Failure - Loose Baffles in Torus

Prepared by: ____________________ ________
T. Young, Jr., Reactor Inspector Date

A. Date and Manner AEC was Informed:

By telephone call from Mr. Tom McCluskey, Station Superintendent, at 11:40 a.m. on May 4, 1972.

B. Description of Particular Event or Circumstance:

An inspection of the torus during the current outage disclosed that five baffles were loose and laying on the floor of the torus; two baffles in the area of one downcomer and three in the area of a second downcomer. The 3/8" bolts that hold the baffles in place were broken.

C. Action by Licensee:

The inspection of the torus is continuing and the licensee is evaluating his findings. A written report will be made to DRL.

8303310563 1p.

APR 24 1972

3944

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

CO INQUIRY REPORT NO. 50-219/72-10
JERSEY CENTRAL POWER & LIGHT COMPANY
OYSTER CREEK - BWR
OTHER - LABOR PROBLEMS RESULTING IN A DELAY OF THE SCHEDULED REFUELING OUTAGE

The subject inquiry report is forwarded for your information.

The full implication of the subject issue is not apparent at this time, but from the information we have received from the licensee, it appears they plan to take a firm stand against the use of the construction craft union to perform the turbine overhaul.

We plan to follow developments closely and will keep you advised 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
CO Files
DR Central Files

B/381

8304150038 1p.

OFFICE ▶	CO					
SURNAME ▶	Cantrell:smg	Carlson				
DATE ▶	4/24/72					

CO Inquiry Report No. 50-219/72-10

Subject: Jersey Central Power & Light Company

License No.: DPR-16

Facility: Oyster Creek - BWR

Title: Other - Labor Problems Resulting in Delay of Scheduled Refueling Outage

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, to the inspector's motel at 9:00 pm on April 21, 1972.

B. Description of Particular Event Or Circumstance:

Mr. McCluskey stated 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 (Inquiry Report 50-219/71-07, October 23, 1971). 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.

C. Action by Licensee:

JCP&L elected to continue operation for another week while labor negotiations continued.

8304150048 1p.

3944

APR 24 1972

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

CO INQUIRY REPORT NO. 50-219/72-09
JERSEY CENTRAL POWER & LIGHT COMPANY
OYSTER CREEK - BWR
EQUIPMENT DEFICIENCY - CRACKS IN SAFETY VALVE SEAT BUSHING

The subject inquiry report is forwarded for your action, since the problem may be generic. We understand that facilities equipped with this type valve include Dresden 2 and 3, Quad Cities 1 and 2, Tsuruga and Nuclenor. (Millstone 1 was originally scheduled to have this type valve; however, they subsequently changed to a combination safety/relief valve - Target Rock.)

Jersey Central appears to be waiting for the results of GE metallurgical analysis prior to making any further comments as to what direction their program will take.

We plan to follow this matter closely and will keep you informed of subsequent developments, as is appropriate.

R. T. Carlson
Senior Reactor Inspector

Enclosure:
Subject Inquiry Report (18 cys)

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

B/382

~~8304180024~~ 1p.

OFFICE ▶	CO					
SURNAME ▶	Cantrell;smg	X Carlson				
DATE ▶	4/24/72					

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

CO Inquiry Report No. 50-219/72-09

Subject: Jersey Central Power & Light Company

License No.: DPR-16

Facility: Oyster Creek - BWR

Title: Equipment Deficiency - Cracks in Safety Valve Seat Bushing

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

A. Date and Manner AEC was Informed:

By the Station Superintendent, Mr. T. J. McCluskey, during a special inspection at the site on April 21, 1972.

B. Description of Particular Event or Circumstance:

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 then 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 No. 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.

~~8304180008~~ 4pp.

The subject 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, Guide F304 stainless steel. The base (or valve housing) is ASTM A216, Guide WCA carbon steel.

C. Action by Licensee:

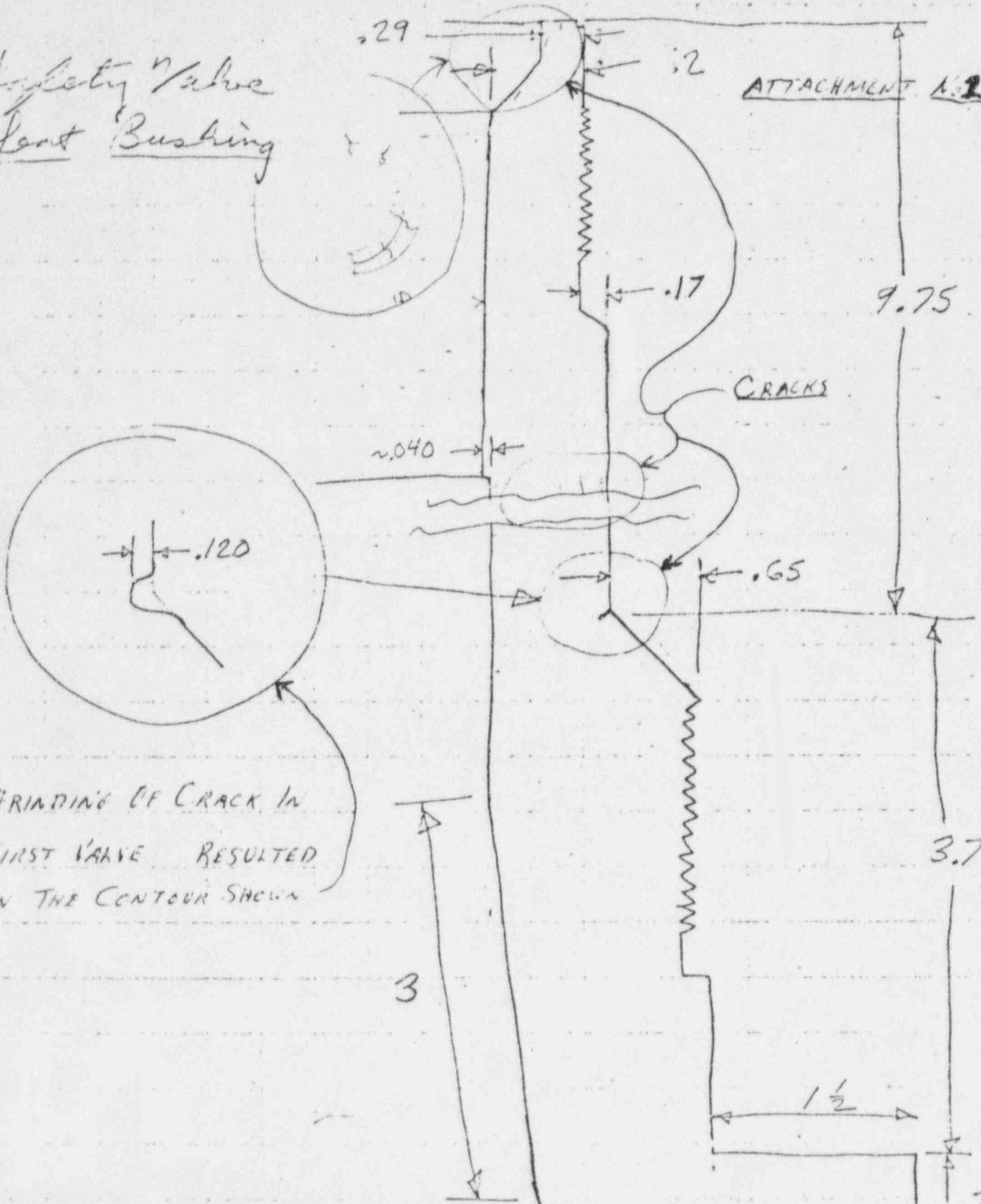
The valve seat of the second valve was shipped to General Electric, San Jose, California by air freight for metallurgical analysis, on April 20, 1972. The results are currently being evaluated by General Electric.

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

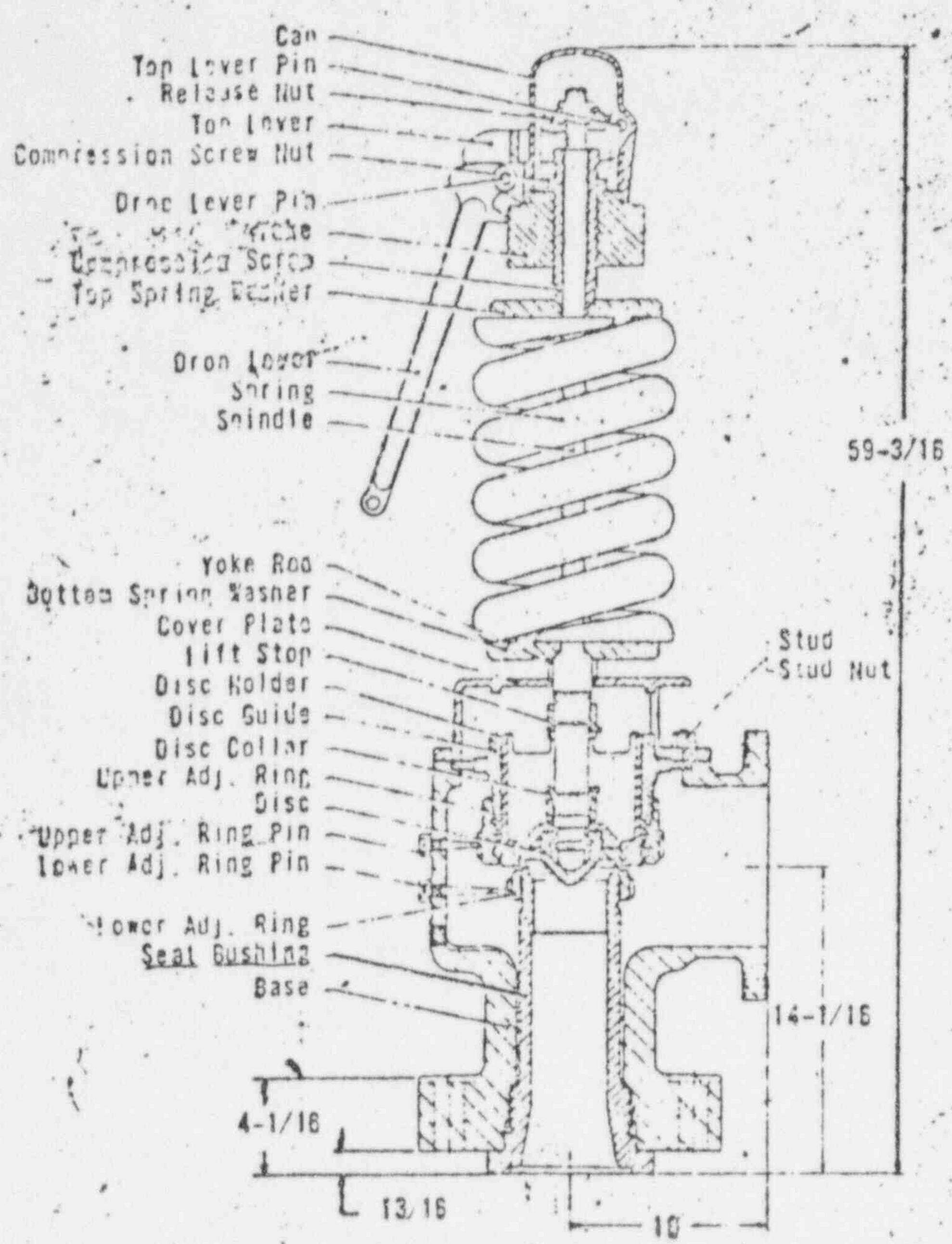
The General Office Review Board (GORB) held a special meeting on April 21, 1972 to review the scheduled turbine trip test in light of the cracks found in the safety valves. The GORB approved the test as scheduled.

Safety Valve
Seat Bushing

ATTACHMENT No. 2



ATTACHMENT No. 2



TSURU

NUCLEHOR

MILLSTONE

DRESDEN 2

QUAD CITIES

GENERAL E
APED -

VPF# 1 9

EP#

Orifice Area

Approximate W

Maximum Press

Maximum Tempe

INLET - 6" - 1500# SPECIAL TONGUE (PER G.E. DRG. 153A2895)

OUTLET - 8" - 150# ASA STANDARD RAISED FACE

Certified:

FORM AEC 755
(7-71)
PI 20013

U.S. ATOMIC ENERGY COMMISSION
COMPLIANCE STATISTISTICAL DATA

A. DOCKET NUMBER	B. REPORT NUMBER	C. PRIORITY/CATEGORY	INQ/INSPECTION/INVESTIGATION DATES FROM D.	TO E.	F. REGION CONDUCTING ACTIVITY
05000219	7203	C	04/21/72		1

LICENSEE VENDOR	FACILITY	LICENSE NUMBER
Jersey Central Power & Light Company	Oyster Creek	DPR-16

G. ACTIVITY CONDUCTED
1 XXX INSPECTION 2 ☐ INQUIRY 3 ☐ INVESTIGATION 4 ☐ VENDER INSPECTION 5 ☐ MANAGEMENT VISIT 6 ☐ INQUIRY—NON LICENSEE

H. INSPECTION/INVESTIGATION RESULTS
1 ☐ 591 2 XXX REGIONAL OFFICE LETTER 3 ☐ REFERRED TO HQS FOR ACTION

J. INSPECTION/INVESTIGATION FINDINGS
1 XXX CLEAR 2 ☐ SAFETY ITEM 3 ☐ NONCOMPLIANCE 4 ☐ NONCONFORMANCE

K. FIELD ACTION AS A RESULT OF INQUIRY
1 ☐ CONDUCT INVESTIGATION 2 ☐ REVIEW NEXT INSPECTION 3 ☐ REFER TO OTHER REGION 4 ☐ REFER TO NON-REG. AUTH. 5 ☐ REFER TO OTHER REG. OFFICE 6 ☐ HQS FOR ACTION 7 ☐ NO FURTHER ACTION

L. REASON INSP. FINDINGS REFERRED TO HEADQUARTERS FOR ACTION
01 ☐ IMMEDIATE THREAT TO HEALTH AND SAFETY
COMPLEX ITEM INVOLVING:
02 ☐ NONCOMPLIANCE/NONCONFORMANCE
03 ☐ LICENSING PROBLEM
04 ☐ POLICY MATTER
05 ☐ INTERPRETATION
06 ☐ SAFETY ITEM
07 ☐ MANAGEMENT DEFICIENCY
08 ☐ INADEQ. REPLY TO LETTER
09 ☐ NO REPLY TO LETTER
10 ☐ NO CORRECTIVE ACTION PLANNED
11 ☐ INADEQUATE CORRECTIVE ACTION PLANNED
12 ☐ HQS LETTER REQUIRED
13 ☐ HQS REVIEW REQUIRED
14 ☐ UNREVIEWED SAFETY MATTER
15 ☐ DESIGN CHANGE
16 ☐ OTHER
17 ☐
18 ☐
19 ☐

M. SUBJECT OF INQUIRY OR INVESTIGATION
01 ☐ TYPE A INT. OVEREXPOSURE
02 ☐ TYPE A EXT. OVEREXPOSURE
03 ☐ TYPE A RELEASE
04 ☐ TYPE A LOSS OF FACILITY
05 ☐ TYPE A PROPERTY DAMAGE
06 ☐ TYPE B INT. OVEREXPOSURE
07 ☐ TYPE B EXT. OVEREXPOSURE
08 ☐ TYPE B RELEASE
09 ☐ TYPE B LOSS OF FACILITY
10 ☐ TYPE B PROPERTY DAMAGE
10 CFR 20.405
11 ☐ INTERNAL OVEREXPOSURE
12 ☐ EXTERNAL OVEREXPOSURE
13 ☐ EXCESSIVE RADIATION LEVELS
14 ☐ EXCESSIVE CONCENTRATION LEVELS
15 ☐ CRITICALITY
16 ☐ LOSS OR THEFT
17 ☐ CONTAMINATION
18 ☐ UNSAFE OPERATION
19 ☐ FIRE, EXPLOSION
20 ☐ HUMAN (OPERATOR) ERROR
21 ☐ COMPLAINT
22 ☐ PUBLIC INTEREST
23 ☐ LEAKING SOURCE
24 ☐ TRANSPORTATION
25 ☐ EXPIRED LICENSE
26 ☐ EXPOSURE REPORTED AND FOUND INVALID
27 ☐ CONSTRUCTION/EQUIP. DEFICIENCY
28 ☐ EQUIPMENT FAILURE
29 ☐ EXCEED LIC/TECH SPEC REQ'S
30 ☐ DEPARTURE FROM FSAR/TS'S
31 ☐ OTHER

N. HEADQUARTERS ACTION ON INSPECTION AND INVESTIGATION
01 ☐ NO ACTION REQUIRED
02 ☐ LETTER-CLEAR
03 ☐ LETTER-NONCOMPLIANCE
04 ☐ LETTER-SAFETY ITEM
05 ☐ PART 2 NOTICE
06 ☐ PART 2 NOTICE AS RESULT OF FOLLOWUP TO REGIONAL OFFICE LETTER
07 ☐ ORDER
08 ☐ REFER TO DRL FOR RESOLUTION
09 ☐ REFER TO DRL FOR INFORMATION
10 ☐ REFER TO DML FOR RESOLUTION
11 ☐ REFER TO DML FOR INFORMATION
12 ☐ REFER TO REGION TO CLOSE OUT
13 ☐ OTHER

REGIONAL OFFICE ACTION DATES
O. REPORT SENT TO HEADQUARTERS: 05-25-72
P. 591/LETTER ISSUED: 05-25-72
Q. X REPLY NOT REQUIRED
R. LICENSEE REPLY RECEIVED
S. ☐ REPLY INADEQUATE

T. DATE LETTER, NOTICE, ORDER ISSUED
U. DATE LICENSEE REPLY RECEIVED
V. ☐ REPLY NOT REQUIRED

B/383