ENCLOSURE NO. 1

Description of Violations

Jersey Central Power & Light Company
Madison Avenue at Punch Bowl Road
Morristown, New Jersey 07960
License No. DPR-16
Docket No. 50-219

Certain activities under your license appear to be in violation of AEC requirements.

A. These apparent violations are considered to be of Category II Severity.

1. 10 CFR 20.201(b) specified that "Each licensee shall make or cause to be made such surveys as may be necessary for him to comply with the regulations in this part." 10 CFR 20.101(b) limits whole body exposure to three rem/quarter.

   Contrary to the above requirements, exposure records show that one man was exposed in excess of three rem during the April-June 1973 quarter.

   It was noted that additional controls over contractor employees have been initiated in order to prevent a recurrence.

2. Paragraph 4.6.C of the Technical Specification requires that a reactor coolant sample be analyzed every 72 hours for total radioactive iodine content.

   Contrary to the above requirement, this analysis was not made for approximately 104 hours between June 9 and June 13, 1973.

   We note that additional controls have been initiated to prevent a recurrence.

B. This apparent violation is considered to be of Category III severity.

1. Paragraph 4.2 of the Technical Specification requires that the control rod drive housing support system be inspected after reassembly.

Paragraph 6.5 of the T.S. requires "Records of principal maintenance activities including inspection and repair, of principal items of equipment pertaining to nuclear safety."

Contrary to the above requirement, the reactor was restarted following reassembly of the control rod drive support system without any records of the required inspection. We note that to prevent recurrence the drywell closure check-off sheet (procedure 202.4) has been revised to provide written verification of support system inspection (when required) prior to closure of the drywell.

Jersey Central Power & Light Company

MADISON AVENUE AT PUNCH BOWL ROAD • MORRISTOWN, N.J. 07960 • 539-6111

September 21, 1973

Mr. A. Giambusso
Deputy Director for Reactor Projects
Directorate of Licensing
United States Atomic Energy Commission
Washington, D. C. 20545

Dear Mr. Giambusso:

Subject: Oyster Creek Station
Docket No. 50-219
Hydraulic Shock and Sway Arrestor Failure

The purpose of this letter is to provide information pertaining to additional hydraulic shock and sway arrestor failures at the Oyster Creek station. A written report as requested in R. O Bulletin No. 73-4 will be submitted upon completion of our inspection program.

This event is considered to be an abnormal occurrence as defined in the Technical Specifications, paragraph 1.15.D. Notification of this event as required by the Technical Specifications, paragraph 6.6.2.a., was made to AEC Region I, Directorate of Regulatory Operations, by telephone on September 10, 1973, and in writing to Mr. B. Greenman on September 11, 1973, during his visit to the site.

As committed to in a letter from Mr. D. A. Ross to Mr. A. Giambusso dated July 27, 1973, the plant was shut down on September 8, 1973 for the purpose of inspecting the hydraulic shock and sway arrestors located on piping systems throughout the drywell and reactor building. Partial inspection has revealed the failure of 23 out of 66 hydraulic shock and sway arrestors on piping systems in the drywell and 19 out of 72 (so far inspected) external to the drywell, some of which are associated with engineered safeguards systems. All 25 Grinnell shock absorbers on the five recirculation loops in the drywell were inspected and found to contain varying amounts of oil. None, however, were without oil and, consequently, are considered to be operable. They have been refilled as necessary under normal maintenance.

The snubbers were made inoperable due to excessive loss of hydraulic fluid resulting from the failed millable gum polyurethane seals.

The failed hydraulic shock and sway arrestors are being, or will be, replaced using snubbers rebuilt with seal kits supplied by the Bergen-Paterson Pipe Support Company. Further, all the remaining units in the drywell, although

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noted to be satisfactory during this inspection, are being removed and rebuilt to replace certain critical millable gum polyurethane seals with molded polyurethane seals in the kits noted above. The seals provided to us by Bergen Paterson should provide a longer service life than those previously utilized at Oyster Creek.

The failures were such that they affected both core spray and both emergency condenser systems in the drywell and both containment spray systems external to the drywell. The loss of shock absorber operability results in a reduction in the ability of the associated piping systems to withstand a design bases earthquake. Additionally, failed shock absorber restraints were also discovered on the electromatic relief discharge blowdown lines. Failure of these shock absorbers would increase the probability of damage to the relief valve discharge piping during periods of multiple valve actuation.

The following program is proposed which is intended to provide a permanent modification of our hydraulic shock absorber units to assure their proper long term operation:

We will conduct our next reinspection of our Bergen-Paterson hydraulic shock absorbers following approximately 6 weeks and no longer than 12 weeks after the plant has been at operating temperature following this shutdown. N.G.

The inspection will include all the items as identified in R. O. Bulletin No. 73-4 dated August 17, 1973. At the time of the reinspection, a prompt telephone report will be made to advise Region I of our findings. A written report will also be provided as specified in the above mentioned R. O. Bulletin.

Due to the significant radiation exposure associated with the inspection and complete rebuild of all our Bergen-Paterson shock absorbers (approximately 20 man-rem to date for this inspection and repair), we believe it most desirable to have the reinspection period coincide with the availability of an ultimate modification, if at all possible.

Our Generation Engineering Department is currently pursuing two equally acceptable long term solutions. First, General Electric Company has underway a development program with the snubber and seal manufacturer to determine that the permanent satisfactory seal will be of ethylene propylene material. Once this is determined and agreed to be all parties involved, we would purchase enough seal kits to rebuild all our Bergen-Paterson shock absorbers or consider purchasing new Bergen-Paterson units, utilizing the ethylene propylene seal material for installation in the drywell.

A second solution currently being pursued is the consideration of replacing all the Bergen-Paterson shock absorbers presently in the drywell with those of a different manufacturer.

Selection of either alternate will be considered in view of the long term suitability to resolve the hydraulic shock failure problem, timely availability of material and minimum exposure to station personnel implementing the repair.

We will keep the regional office advised of our progress in arriving at a timely long range solution.

We are enclosing forty copies of this report.

Very truly yours,

Donald A. Ross

Donald A. Ross
Manager, Nuclear Generating Stations

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Enclosures

cc: Mr. J. P. O'Reilly, Director
Directorate of Regulatory Operations, Region I

Jersey Central Power & Light Company

MADISON AVENUE AT PUNCH BOWL ROAD • MORRISTOWN, N. J. 07960 • 539-6111

September 21, 1973

AO-73-24

Mr. A. Giambusso
Deputy Director for Reactor Projects
Directorate of Licensing
United States Atomic Energy Commission
Washington, D. C. 20545

Dear Mr. Giambusso:

Subject: Oyster Creek Station
Docket No. 50-219
Failure of Main Steam Isolation Valve

The purpose of this letter is to report a failure of the main steam isolation valve NS03B to meet the acceptable leakage rate criterion as specified in Technical Specifications 4.5.F.1.D. This event is considered a violation of the Technical Specifications, paragraph 1.15.E.

This event is also considered to be an abnormal occurrence as defined in the Technical Specifications, paragraph 1.15.E. Notification of this event as required by the Technical Specifications, paragraph 6.6.2.a., was made to AEC Region I, Directorate of Regulatory Operations, by telephone on September 10, 1973 and personally to Mr. E. Greenman on September 10, 1973.

The reactor was shut down on September 8, 1973 for the purpose of re-inspecting the Bergen-Paterson shock absorbers at the Oyster Creek station. A leakage rate test was conducted on the main steam isolation valves in accordance with previous commitments to the Atomic Energy Commission. As a result of this testing, which is partially completed, the leakage rate for NS03B was found to be approximately 200 SCFH based on the rate of pressure buildup between valves NS03B and NS04B. The allowable leakage rate limit, as detailed in the Technical Specifications, is 9.95 SCFH (5% of L_{to} [20]). The other inside valve NS03A leakage rate was determined to be nondetectable, <0.1 SCFH. The leakage measurements for the outside isolation valves will be determined once we have completed the current inspection and repair of NS03B.

This failure is similar to one reported to your office by my letter dated June 5, 1973. As a result of the failure to achieve an acceptable leakage rate measurement at that time, we disassembled NS03B, the pilot stem was removed and replaced with a new one manufactured to new specifications and close quality controls. In addition, both the main seat and pilot seat surfaces were relapped. Following reassembly of the valve, the 20 psi air test indicated no detectable leakage through the valve (i.e., <0.1 SCFH). It was believed at that time that

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the failure of NS03B to pass the air test was due to the lack of straightness in the original pilot stem. The original stem, on several previous occasions, was straightened and reinstalled in the valve and acceptable leakage test subsequently performed on the valve. However, based on our investigation into the recurring leakage problems with this particular valve, it was judged that the stem was relaxing after operating for a period of time at elevated temperatures, resulting in excessive stem bowing and improper pilot valve seating. Therefore, two replacement stems were manufactured by Atwood & Morrill Company to special specifications provided by Jersey Central Power & Light Company.

The failure which is being reported by this letter reflects the results of the first test subsequent to some operating history on NS03B with the specially manufactured pilot stem.

The cause of the excessive pilot leakage is unknown at this time.

A special meeting was held at the Oyster Creek station on September 13, 1973 to review the most recent developments with this particular valve.

The following course of action was agreed upon by Jersey Central Power & Light Company, Atwood & Morrill Company and two other companies consulting with Jersey Central Power & Light Company on this problem:

A. Prior to Disassembling of NS03B

1. Instrument with dial gauge and potentiometer to measure stem stroke at valve closure. Obtain baseline marks before operating valve.

2. Instrument cylinder to measure Δp across cylinder.

3. Perform stroke tests, measure cylinder Δp and valve stroke repeatability. As a part of this, also measure stem movement at a junction of cylinder Δp for increments from $\Delta p = 0$ to $\Delta p =$ design. Also check packing friction by loosening and checking stem motion and repeatability.

4. Determine whether stem is installed such that it is not "bottoming-out" on top or bottom of operator cylinder.

5. Check runout of coupling between valve and operator stem.

NOTE: If measurements indicate significant changes during the stroke tests, conduct leakage tests to determine effect on valve leakage.

B. After Disassembly of NS03B

1. Perform complete dimensional inspection of critical valve parts.

2. Cylinder examination for obstructions, rust, etc., and condition of seals.

3. Refurbish, as required, (Atwood & Morrill indicated they can provide qualified welders and procedures, if required, for restelliting guide surfaces.)

4. Repeat stem stroke repeatability and Δp measurements, note above, after refurbishment.

C. Long Term Action

It was agreed that a better lapping tool with bearings and internal supports is needed. Atwood & Morrill indicated that such a tool is being developed by them and is expected to be available in October 1973. Atwood & Morrill will advise Jersey Central Power & Light Company of the schedule for delivery of a lapping fixture for the Oyster Creek valves.

D. Procedures

The inspections and examinations outlined above will be performed under Atwood & Morrill's and Jersey Central Power & Light Company's supervision in accordance with written procedures. These procedures are the responsibility of Jersey Central Power & Light Company and will be reviewed by General Electric and Atwood & Morrill.

In addition, Atwood & Morrill will also furnish a representative to follow this work.

In determining the significants of this valve leakage, the rate of pressure buildup in the reactor was compared to a graph of pressure buildup where at least one valve in each steam line was leak tight. These plots compared favorably. This implies that one valve in the "B" main steam line (i.e., NS04B) is leak tight. This was confirmed when pressure buildup between the valves was observed to be approximately the same as the reactor pressure. The redundancy feature will be confirmed upon successful completion of the NS04B leak test.

It is not possible, at this time, to specify exactly what corrective actions are to be taken to prevent the reoccurrence of this situation. The course of action will be dictated upon completion of the analysis of the extensive dimensional inspection described above. It is our intention to keep your office informed informally through our Region I compliance inspector; and, following the completion of the program described herein, to forward to your office the written results of our inspection and the corrective actions dictated by this inspection.

We are enclosing forty copies of this report.

Very truly yours,

Donald A. Ross
Manager, Nuclear Generating Stations

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Enclosures

cc: Mr. J. P. O'Reilly, Director
Directorate of Regulatory Operations, Region I