


50-219

Jersey Central Power & Light Company

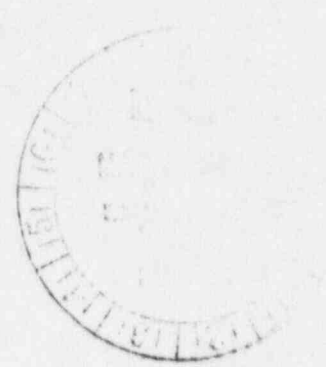


MADISON AVENUE AT PUNCH BOWL ROAD • MORRISTOWN, N. J. 07960 • 201-539-6111

MEMBER OF THE
General  Public Utilities Corporation

March 18, 1974

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
Abnormal Occurrence Report No. 50-219/74/18

The purpose of this letter is to forward to you the attached Abnormal Occurrence Report in compliance with paragraph 6.6.2.a of the Technical Specifications.

Enclosed are forty copies of this submittal.

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

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OYSTER CREEK NUCLEAR GENERATING STATION
FORKED RIVER, NEW JERSEY 08731

Abnormal Occurrence
Report No. 50-219/74/18

Report Date

March 18, 1974

Occurrence Date

March 8, 1974

Identification of Occurrence

Inoperability of one (1) Bergen-Paterson hydraulic shock and sway arrestor located on the "A" isolation condenser in the drywell. This event is considered to be an abnormal occurrence as defined in the Technical Specifications, paragraph 1.15D.

Conditions Prior to Occurrence

The plant was shut down with reactor coolant at <212°F.

Description of Occurrence

An inspection of the drywell snubbers, Bergen-Paterson type HSSA-10, located three inoperable units and four which were leaking. They are as follows:

<u>Unit</u>	<u>System</u>	<u>Condition</u>	<u>Elev.</u>
F93501 #2	A Isolation Condenser	Inoperable	75'
487574	B Isolation Condenser	Leaking	95'
487502	B Isolation Condenser	Leaking	95'
487495	Cleanup System	Leaking	55'
487573	Shutdown Cooling	Inoperable	51'
487489	North Electromatic Relief	Leaking	51'
487446	South Electromatic Relief	Inoperable	51'

All of the above HSSA units were rebuilt in January 1974 with ethylene propylene seals, except Unit #487489. This unit was rebuilt primarily with molded polyurethane seals in September 1973.

Of the three inoperable units, only one (F93501 #2) is associated with an engineered safeguard system ("A" Isolation Condenser). It is not believed that temperatures above the normal ambient conditions for the drywell during operation existed in the vicinity of any of the three inoperable units.

Apparent Cause of Occurrence

The cause of snubber inoperability was a loss of the hydraulic fluid. An investigation is being initiated to determine why the fluid was expelled.

Analysis of Occurrence

The safety significance of this occurrence was a partial loss of the seismic restraining ability for the affected systems. Had the plant suffered a design basis earthquake, the probability that these systems would have suffered structural damage was increased.

Corrective Action

The seven units were replaced with identical snubbers which were rebuilt with ethylene propylene seals. In addition, pressure tests will be run on the defective units in an attempt to determine the cause of their inoperability and then each of the units will be dismantled to visually determine seal status. The results of this program will provide additional information toward a resolution of this problem.

Failure Data

Manufacturer: Bergen-Paterson
Type: HSSA-10


Jersey Central Power & Light Company



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March 18, 1974

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
Abnormal Occurrence Report No. 50-219/74/19

The purpose of this letter is to forward to you the attached Abnormal Occurrence Report in compliance with paragraph 6.6.2.a of the Technical Specifications.

Enclosed are forty copies of this submittal.

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

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OYSTER CREEK NUCLEAR GENERATING STATION
FORKED RIVER, NEW JERSEY 08731

Abnormal Occurrence
Report No. 50-219/74/19

Report Date

March 18, 1974

Occurrence Date

March 9, 1974

Identification of Occurrence

Violation of the Technical Specifications, paragraph 3.5.A.1, failure to maintain primary containment integrity with reactor water temperature above 212°F and fuel in the reactor vessel. This event is considered to be an abnormal occurrence as defined in the Technical Specifications, paragraph 1.15B.

Conditions Prior to Occurrence

The plant was shut down with reactor coolant <212°F.

Description of Occurrence

At 1000 on March 8, 1974, an orderly shutdown of the plant commenced to perform maintenance on six of the fourteen torus to drywell vacuum breaker valves. Although thirteen of the valves were considered to be operable at this time, the plant was shut down in order to effect more permanent repairs on the valves. (See Abnormal Occurrence Report No. 50-219/74/16 dated March 15, 1974). In accordance with the requirements of paragraph 3.5.A.1 of the Technical Specifications, maintenance on these valves did not begin until reactor coolant temperature was below 212°F.

At approximately 0500 on March 9, 1974, the control room operator noticed that the reactor vessel head metal temperature was increasing at a rate of 10°F/hr after being stable at 231°F for about a 3-hour period. A check of the recirculation loop temperature recorder, which was being used as the primary indication of reactor water temperature, showed the recorder trace to be constant at 160°F. While this inconsistency was being investigated, a radiation protection technician reported to the shift foreman that water vapor appeared to be issuing from a special manometer which had been installed for monitoring of the pressure difference between the drywell and reactor vessel. The shift foreman's investigation revealed that the reactor side of the manometer was hot. The recirculation loop temperature recorder was then rechecked and again indicated a temperature of 160°F. However, the indication jumped to approximately 250°F when the recorder was bumped. Since two of the torus to drywell vacuum breaker valves were undergoing maintenance at this time (0531 on March 9, 1974), primary

containment integrity was not in effect. Shutdown cooling system flow was then increased to decrease the reactor water temperature. Reactor water temperature was reduced to less than 212°F within approximately 30 minutes. Within approximately 130 minutes, a reactor water temperature of 160°F was established and maintained.

Apparent Cause of Occurrence

This abnormal occurrence is attributed to equipment malfunction and operator error. The recirculation loop recorder did stick and give a false indication of reactor coolant temperature. However, the control room operator failed to react properly to indications that reactor water temperature was increasing. Specifically, a review of the chart paper from the recorder monitoring shutdown cooling system temperatures showed that the "C" loop heat exchanger inlet temperature was increasing at a rate of about 10°F/hr during the 3-hour period prior to 0230 on March 9, 1974. At this time, the control room operator secured flow in this loop and thereby contributed to the rise in reactor water temperature.

Analysis of Occurrence

The primary containment system provides a barrier against uncontrolled release of fission products to the environs in the event of a break in the reactor coolant systems. Whenever the reactor coolant water temperature is above 212°F, failure of the reactor coolant system could cause rapid expulsion of the coolant from the reactor with an associated pressure rise in the primary containment. Primary containment is required, therefore, to contain the thermal energy of the expelled coolant and fission products which would be released from any fuel failures resulting from the accident.

The safety significance of this event is that primary containment integrity was not maintained during the period that the reactor coolant temperature was in excess of 212°F due to the maintenance being performed on the vacuum breaker valves. At the condition that existed, the safety significance is considered minimal.

Corrective Action

The following corrective actions will be taken:

1. The operator retraining program will emphasize that all available indications are to be used when critical parameters such as reactor water temperature are being monitored. In addition, the operators will be instructed in their retraining to regard extremely straight recorder traces as suspect and to view confirming indications whenever such suspicious recorder traces are encountered, both while operating and shut down. ✓

2. The shutdown log will be reviewed and modified to require the recording of additional system temperatures related to the reactor coolant temperature.

Failure Data

Basic recorder data are as follows:

Manufacturer: General Electric Company
Type: GE/MAC 531
Span: 4 inches