

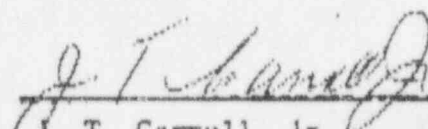
To: James P. O'Reilly
Directorate of Regulatory Operations
Region I
631 Park Avenue
King of Prussia, Pennsylvania 19406

From: Jersey Central Power & Light Company
Oyster Creek Nuclear Generating Station Docket #50-219
Porked River, New Jersey 08731

Subject: Abnormal Occurrence Report No. 50-219/74/ 20

The following is a preliminary report being submitted
in compliance with the Technical Specifications
paragraph 6.6.2.

Preliminary Approval:


J. T. Carroll, Jr. 3/11/74
Date

cc: Mr. A. Giambusso

8/598

Initial Telephone
Report Date: 3/11/74

Date of
Occurrence: 3/10/74

Initial Written
Report Date: 3/11/74

Time of
Occurrence: 1239

OYSTER CREEK NUCLEAR GENERATING STATION
FORKED RIVER, NEW JERSEY 08731

Abnormal Occurrence
Report No. 50-219/74/20

IDENTIFICATION
OF OCCURRENCE:

Violation of the Technical Specifications, paragraph 4.5.4.1.d,
failure of Main Steam Isolation Valves NS04A and NS04B to meet
the allowable leakage requirements.

This event is considered to be an abnormal occurrence as de-
fined in the Technical Specifications, paragraph 1.15D.8.E.

CONDITIONS PRIOR
TO OCCURRENCE:

<input type="checkbox"/> Steady State Power	<input type="checkbox"/> Routine Shutdown
<input type="checkbox"/> Hot Standby	<input type="checkbox"/> Operation
<input type="checkbox"/> Cold Shutdown	<input type="checkbox"/> Load Changes During
<input type="checkbox"/> Refueling Shutdown	<input type="checkbox"/> Routine Power Operation
<input type="checkbox"/> Routine Startup	<input checked="" type="checkbox"/> Other (Specify)
<input type="checkbox"/> Operation	<input type="checkbox"/> See below

The plant was shutdown with the reactor coolant at $<212^{\circ}\text{F}$,
with the reactor mode in REFUEL.

DESCRIPTION
OF OCCURRENCE:

The MSIV's were tested in the "as found" condition to the
extent that the valves weren't cycled before the test. The
valves were not, however, closed under pressure.

1159 - Leak rate tests on MSIV's NS04A and NS04B began

1239 - Leak rate tests on MSIV's NS04A and NS04B ended

Leakage rates of MSIV's NS04A and NS04B were 64.7 SCFH and 12.2 SCFH, corrected to 20 psi. The maximum allowable leakage rate is 9.945 SCFH, as required by the Technical Specifications, paragraph 4.5.F.1.d.

APPARENT CAUSE
OF OCCURRENCE:

<input type="checkbox"/>	Design	<input type="checkbox"/>	Procedure
<input type="checkbox"/>	Manufacture	<input type="checkbox"/>	Unusual Service Condition
<input type="checkbox"/>	Installation/ Construction	<input type="checkbox"/>	Inc. Environmental
<input type="checkbox"/>	Operator	<input checked="" type="checkbox"/>	Component Failure
		<input type="checkbox"/>	Other (Specify)

After checking the test assembly and the components of the MSIV's, it was determined that the lower packing ring around the valve shaft was the cause of the excessive leak rate. The leakage was out of the leakoff line between the upper and lower sets of packing. A cause of repetitive leakage may be that, when the valve is repacked in place, the packing ring is cut and then installed around the shaft. This packing is designed for installation by sliding the packing rings down the shaft without cutting. The valve operator must be disconnected for this method of installation.

ANALYSIS OF
OCCURRENCE:

The safety significance of the failure of NS04A and NS04B to pass the leakage rate test was a loss of redundancy in an engineered safety feature designed to minimize the release of fission products under design bases accident conditions. It should be noted that any leakage through the lower set of packing would be into the reactor building equipment drain tank

and would be released through the plant stack via the standby gas treatment system. It should also be noted that the inside MSIV's are reported to have no detectable leakage and, therefore, there would be no leakage out of the primary containment.

CORRECTIVE ACTION:

The MSIV valve shaft packing leakoff valve (between upper and lower sets of packing) were closed and the MSIV's were retested successfully. These valves will remain in the closed position until the 1974 refueling outage, at which time the valve shafts will be repacked. NS03A and NS03B were retested to insure that they had an acceptable leak rate, since the test assumes that the valves, NS04A and NS04B, have negligible leakage. The retests of NS03A and NS03B indicated no detectable leakage.

FAILURE DATA:

The valve stem packing on NS04A failed on September 27, 1973 and again on January 16, 1974. Each time, the valve was repacked and subsequently passed its leak rate test. The valve stem packing on NS04B failed on September 27, 1973 and was subsequently repacked.

Prepared by: *E. J. [Signature]* Date: 3/11/74

To:

James P. O'Reilly
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From:

Jersey Central Power & Light Company
Oyster Creek Nuclear Generating Station Docket #50-219
Forked River, New Jersey 08731

Subject:

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

The following is a preliminary report being submitted
in compliance with the Technical Specifications
paragraph 6.6.2.

Preliminary Approval:

J. T. Carroll, Jr. 3/11/74
J. T. Carroll, Jr. Date

cc: Mr. A. Giambusso

~~834414418~~ (5R)

B/599

OYSTER CREEK NUCLEAR GENERATING STATION
FORKED RIVER, NEW JERSEY 08731

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

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 de-
fined in the Technical Specifications, paragraph 1.15B.

CONDITIONS PRIOR
TO OCCURRENCE:

- | | |
|---|--|
| <input type="checkbox"/> Steady State Power | <input type="checkbox"/> Routine Shutdown |
| <input type="checkbox"/> Hot Standby | <input type="checkbox"/> Operation |
| <input type="checkbox"/> Cold Shutdown | <input type="checkbox"/> Load Changes During |
| <input type="checkbox"/> Refueling Shutdown | <input type="checkbox"/> Routine Power Operation |
| <input type="checkbox"/> Routine Startup | <input type="checkbox"/> Other (Specify) |
| <input type="checkbox"/> Operation | |

The plant was shutdown 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 shutdown in order to effect more permanent repairs on the
valves (see Abnormal Occurrence Report No. 74-16, dated

March 8, 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 0520 on March 9, 1974, 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, which was being used to monitor reactor water temperature, was immediately rechecked. This recorder indicated a temperature of 160°F. However, the indication jumped to approximately 250°F when the recorder was bumped. At this time (0531 on March 9, 1974) shutdown cooling system flow was 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 180°F was established and maintained.

APPARENT CAUSE
OF OCCURRENCE:

<input checked="" type="checkbox"/>	Design	<input type="checkbox"/>	Procedure
<input type="checkbox"/>	Manufacture	<input type="checkbox"/>	Unusual Service Condition
<input type="checkbox"/>	Installation/	<input type="checkbox"/>	Inc. Environmental
<input type="checkbox"/>	Construction	<input type="checkbox"/>	Component Failure
<input checked="" type="checkbox"/>	Operator	<input type="checkbox"/>	Other (Specify)

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 three hour period prior to 0230 on March 9, 1973. 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 remedial actions will be taken prior to the PORC evaluation to preclude a recurrence of this type event:

1. Control room operators will be instructed to "jog" any recorder that is producing a suspiciously straight trace. This will be accomplished by momentarily turning the recorder off and then on again.
2. Involved personnel will be reminded to utilize all available indicators when monitoring critical parameters such as reactor water temperature.
3. The shutdown log will be reviewed and modified to require the recording of additional system temperatures which are related to the reactor coolant temperature.

FAILURE DATA:

Basic recorder data are as follows:

Manufacturer - General Electric
Type - GE/MAC 531
Span - 4 inches

Prepared by:



Date:

3/11/74

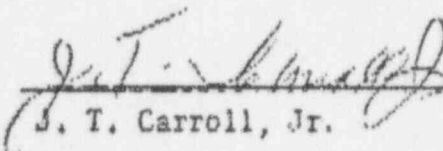
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Subject: Abnormal Occurrence Report No. 50-219/74/ 18

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Preliminary Approval:


J. T. Carroll, Jr. 3/11/74
Date

cc: Mr. A. Giambusso

~~81-02231-1 (300)~~

B/600

3/11/74 Occurrence: 1800
OYSTER CREEK NUCLEAR GENERATING STATION
FORKED RIVER, NEW JERSEY 08731

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

IDENTIFICATION
(OF OCCURRENCE):

Violation of the Technical Specifications, paragraph N/A,
Failure of Bergen-Paterson Hydraulic Shock and Sway Arrestors
in the drywell.

This event is considered to be an abnormal occurrence as de-
fined in the Technical Specifications, paragraph 1.15D.

CONDITIONS PRIOR
TO OCCURRENCE:

<input type="checkbox"/>	Steady State Power	<input type="checkbox"/>	Routine Shutdown
<input type="checkbox"/>	Hot Standby	<input type="checkbox"/>	Operation
<input checked="" type="checkbox"/>	Cold Shutdown	<input type="checkbox"/>	Load Changes During
<input type="checkbox"/>	Refueling Shutdown	<input type="checkbox"/>	Routine Power Operation
<input type="checkbox"/>	Routine Startup	<input type="checkbox"/>	Other (Specify)
<input type="checkbox"/>	Operation		

The plant was shutdown 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:

F93501 #2	- A Isolation Condenser	- Failed
487574	- B Isolation Condenser	- Leaking
487502	- B Isolation Condenser	- Leaking
487495	- Cleanup System	- Leaking
487573	- Shutdown Cooling	- Failed
487489	- North Electromatic Relief	- Leaking
487446	- South Electromatic Relief	- Failed

All of the above units, with the exception of 487489, were
rebuilt in January 1974 with EP seals.

APPARENT CAUSE
OF OCCURRENCE:

- Design
- Manufacture
- Installation/
- Construction
- Operation

- Pro Jure
- Unusual Service Condition
- Inc. Environmental
- Component Failure
- Other (Specify)

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 bases earthquake, the probability that these systems would have suffered structural damage was increased.

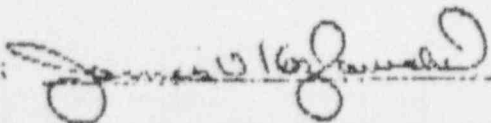
CORRECTIVE
ACTION:

The failed units were replaced with identical snubbers which were rebuilt with ethylene propylene seals.

FAILURE DATA:

Manufacturer: Bergen-Paterson
Type: HSSA-10

Prepared by:



Date:

3/11/74