NRC FORM 366 (7-77) U.S. NUCLEAR REGULATORY COMMISSION

# LICENSEE EVENT REPORT

CONTROL BLOCK: | | | | | | | (1) (PLEASE PRINT OR TYPE ALL REQUIRED INFORMATION)

0 1 | N J O C P 1 (2) | 0 0 - 0 0 0 0 0 - 0 0 (3) | 4 1 1 1 1 (4) | | (5)
LICENSEE CODE | LICENSE NUMBER | LICENSE TYPE | CAT

CON'T 0 1 REPORT SOURCE | L (6) | 0 5 0 0 0 2 1 9 (7) | 1 1 2 1 7 8 (8) | 1 2 0 5 7 8 (9)
DOCKET NUMBER | EVENT DATE | REPORT DATE

EVENT DESCRIPTION AND PROBABLE CONSEQUENCES (10)

0 2 On November 21, 1978, during the performance of an integrated primary
0 3 containment leak rate test, the torus to reactor building differential
0 4 pressure switches, DPS-66A & B, were found to have ruptured diaphragms.
0 5 This condition was found while looking for suspected sources of leakage
0 6 during the performance of the test. After closing the instrument root
0 7 valves for the failed switches, the total leakage attributable to the
0 8 ruptured diaphragms was calculated to be approximately 3 SCFM.

| | SYSTEM CODE | CAUSE CODE | CAUSE SUBCODE | COMPONENT CODE | COMP. SUBCODE | VALVE SUBCODE |
|---|---|---|---|---|---|---|
| 0 9 | I B (11) | B (12) | A (13) | I N S T R U (14) | E (15) | Z (16) |

| (17) LER/RO REPORT NUMBER | EVENT YEAR | | SEQUENTIAL REPORT NO. | | OCCURRENCE CODE | REPORT TYPE | | REVISION NO. |
|---|---|---|---|---|---|---|---|---|
| | 7 8 | — | 0 2 6 | / | 0 1 | T | — | 0 |

| ACTION TAKEN | FUTURE ACTION | EFFECT ON PLANT | SHUTDOWN METHOD | HOURS (22) | ATTACHMENT SUBMITTED | NPRD-4 FORM SUB. | PRIME COMP. SUPPLIER | COMPONENT MANUFACTURER |
|---|---|---|---|---|---|---|---|---|
| C (18) | Z (19) | Z (20) | Z (21) | 0 0 0 0 0 | Y (23) | Y (24) | N (25) | D 2 8 2 (26) |

CAUSE DESCRIPTION AND CORRECTIVE ACTIONS (27)

1 0 The failed ΔP switches were not designed to withstand the ΔP to which
1 1 they were subjected. They are rated for a maximum of 10 psid continuous
1 2 and a surge pressure of 25 psid. The failed switches were replaced with a
1 3 new type designed to withstand accident conditions.
1 4

| | FACILITY STATUS | % POWER | OTHER STATUS (30) | METHOD OF DISCOVERY | DISCOVERY DESCRIPTION (32) |
|---|---|---|---|---|---|
| 1 5 | H (28) | 0 0 0 (29) | NA | B (31) | Primary Containment ILRT |

| | ACTIVITY RELEASED | CONTENT OF RELEASE | AMOUNT OF ACTIVITY (35) | LOCATION OF RELEASE (36) |
|---|---|---|---|---|
| 1 6 | Z (33) | Z (34) | NA | NA |

PERSONNEL EXPOSURES

| | NUMBER | TYPE | DESCRIPTION (39) |
|---|---|---|---|
| 1 7 | 0 0 0 (37) | Z (38) | NA |

PERSONNEL INJURIES

| | NUMBER | DESCRIPTION (41) |
|---|---|---|
| 1 8 | 0 0 0 (40) | NA |

LOSS OF OR DAMAGE TO FACILITY (43)

| | TYPE | DESCRIPTION |
|---|---|---|
| 1 9 | Z (42) | NA |

PUBLICITY

| | ISSUED | DESCRIPTION (45) | NRC USE ONLY |
|---|---|---|---|
| 2 0 | Y (44) | Weekly press release - December 12, 1978. | |

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JCP&L GPU

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OYSTER CREEK NUCLEAR GENERATING STATION
Forked River, New Jersey 08731

Licensee Event Report
Reportable Occurrence No. 50-219/78-26/1T-0

## Report Date

December 5, 1978

## Occurrence Date

November 21, 1978

## Identification of Occurrence

Failure of the torus to reactor building differential pressure switches DPS-66A and 66B. This event is considered to be a reportable occurrence as defined in the Technical Specifications, paragraph 6.9.2.a.3.

## Conditions Prior to Occurrence

The plant was shut down for a scheduled refueling/maintenance outage.

Reactor coolant temperature - 160°F.
Mode switch in "Shutdown".
Reactor vented to the drywell.

## Description of Occurrence

On Tuesday, November 21, 1978, during the performance of an integrated primary containment leak rate test, the torus to reactor building differential pressure switches, DPS-66A and B, were found to have ruptured diaphragms. This condition was discovered while looking for suspected sources of leakage during the performance of the above test. After closing the instrument root valves for these switches, the total leakage attributable to the rupture of the switch diaphragms was calculated to be approximately 3 SCFM.

## Apparent Cause of Occurrence

The cause of this occurrence can be attributed to the fact that the differential pressure switches are rated for a maximum of 10 psid continuous and a surge pressure of 25 psid. Containment pressure at the time of the discovery was 36.6 psia. The differential pressure switches are not designed to withstand the differential pressure to which they were subjected.

Analysis of Occurrence

The rupture of the diaphragms in the differential pressure switches opened a path for the release of atmosphere from the pressure absorption chamber. Had an accident situation occurred and fission products were carried over to the pressure absorption chamber, a release could have occurred from the primary containment to the reactor building. The reactor building is designed to minimize ground level release of airborne radioactive materials and to provide for controlled, elevated release of the building atmosphere under accident conditions. To prevent ground level leakage of fission products from the reactor building, subsequent to design basis accidents, the standby gas treatment system has the capability to maintain a negative pressure of 0.25 inches of water within the reactor building. The system effluent is processed through filters before being discharged through the stack. Leakage from the differential pressure switches would have been processed through the standby gas treatment system and released through the plant stack.

Corrective Action

The differential pressure switches were isolated for the completion of the primary containment integrated leak rate test. The failed switches were replaced with a new type designed to withstand accident conditions. A review was conducted to verify that all instrumentation connected to the primary containment is designed for design basis accident pressure. No other discrepancies were identified.

Failure Data

Manufacturer: Dwyer Instruments, Inc.
Catalog No. 1637-12
0-12 inches of water (set point adjustment) rated at 10 psid.