OYSTER CREEK NUCLEAR GENERATING STATI^N
Forked River, NJ 08731

Licensee Event Report Update
Reportable Occurrence No. 50-219/83-09/01X-2

Report Date

January 22, 1985

Previous Report Date

November 2, 1984

Occurrence Date

February 22, 1983

Identification of Occurrence

The results of local leak rate testing identified ten containment isolation valves and one gasket that failed to meet their acceptance criteria. This constitutes operation of the unit or affected systems when any parameter or operation subject to a limiting condition is less conservative than the least conservative aspect of the limiting condition for operation established in Technical Specifications, paragraph 4.5.F.d.

This event is considered to be reportable as defined in Technical Specification, paragraph 6.9.2.a.2.

Conditions Prior to Occurrence

The plant was in cold shutdown with reactor coolant temperature less than 212^{O}F and the reactor vented at the time the occurrence was identified. The reactor was in various operating modes prior to the occurrence.

Description of Occurrence

Local leak rate testing identified the following ten (10) valves and one (1) gasket with leakage in excess of the acceptance criteria of 12.08 SCFH @20 psig. The results of the leak rate test program for these valves and gasket are as follows:

DESCRIPTION	PENETRATION	DATE TESTED	LEAKAGE @20 PSIG (SCFH)
Instrument Air and Nitrogen System	V-6-395	3/4/83	30.49
MSIV	NS04A	2/14/83	16.34
MSIV	NS04B	2/14/83	17.21
MSIV Drain Valves	V-1-106, 107	2/14/83	19.44
Drywell Headseal	Gasket	2/16/83	544.68
Drywell Purge	V-27-1	2/18/83	34.08
Drywell Sump Discharge	V-22-28, 29	3/17/83	12.4
Drywell Vent	V-27-3, 4	2/27/83	23.19

Apparent Cause of Occurrence

The cause of the leakage is as follows:

I. V-6-395, V-1-106, V-1-107, V-22-28, V-22-29, V-27-3, V-27-4, NS04A, and NS04B had deterioration of valve internals.

II. Drywell Headseal - cause unkown, seal appeared to be in good condition.

III. V-27-1 stem was found to be out of proper alignment.

Analysis of Occurrence

For valves V-27-1, V-6-395, NS04A, NS04B, V-1-106, and V-1-107 at least one redundant valve for each containment penetration met the acceptance criteria.

The purpose of the Containment System is to provide a barrier to limit the release of radioactive material to the environment to less than 10CFR100 limits during design basis accident conditions. The failure of Containment Isolation Valves V-27-3, 4 and the Drywell Head Seal Gasket to meet required acceptance criteria could have resulted in these limits being exceeded. All other individual containment isolation valves which failed leak testing were in series with other redundant isolation valves which did meet the acceptance criteria.

Corrective Action

Valves V-1-106, 107, and V-6-395 have been replaced with new valves. NSO4A and NSO4B had their seats lapped, stems replaced, and packing changed. V-27-1 stem was adjusted. V-22-28 received a new seat, stem, and plug. V-22-29 had its seat lapped. V-27-3, 4 received new seats and the Drywell Head had a new seal installed.

All penetrations passed their subsequent Local Leak Rate Tests.

UPDATE REPORT - PREVIOUS REPORT DATE NOVEMBER 2, 1984

NRC FORM 366 (7-77)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT

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

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

CON'T 01 REPORT SOURCE L (6) 0 5 0 0 0 2 1 9 (7) DOCKET NUMBER 0 2 2 2 8 3 (8) EVENT DATE 0 1 2 2 8 5 (9) REPORT DATE

EVENT DESCRIPTION AND PROBABLE CONSEQUENCES (10)

02 Local leak rate testing revealed that ten containment isolation valves
03 and one gasket failed to meet their acceptance criteria. The failure of
04 Drywell Vent Isolation Valves V-27-3 and 4 and Drywell Head gasket to
05 meet required acceptance criteria could have resulted in exceeding 10CFR
06 100 limits during design basis accident conditions. All other contain-
07 ment isolation valves failing leak testing were in series with redundant
08 valves which passed. Reportable per Tech Specs, paragraph 6.9.2.a.2.

	SYSTEM CODE	CAUSE CODE	CAUSE SUBCODE	COMPONENT CODE	COMP. SUBCODE	VALVE SUBCODE
09	C D (11)	X (12)	Z (13)	V A L V E X (14)	F (15)	D (16)

(17) LER/RO REPORT NUMBER	EVENT YEAR		SEQUENTIAL REPORT NO.		OCCURRENCE CODE	REPORT TYPE		REVISION NO.
	8 3	–	0 0 9	/	0 1	X	–	2

ACTION TAKEN	FUTURE ACTION	EFFECT ON PLANT	SHUTDOWN METHOD	HOURS (22)	ATTACHMENT SUBMITTED	NPRD-4 FORM SUB.	PRIME COMP. SUPPLIER	COMPONENT MANUFACTURER
B (18)	Z (19)	Z (20)	Z (21)	0 0 0 0	Y (23)	N (24)	N (25)	A 5 8

CAUSE DESCRIPTION AND CORRECTIVE ACTIONS (27)

10 Drywell Purge valve V-27-1 stem was out of alignment. Other valves had
11 deterioration of internals. All valves were either repaired or replaced.
12 Although appearing to be in good condition, the Drywell head gasket was
13 replaced. All penetrations passed their subsequent local leak rate
14 tests.

15 FACILITY STATUS H (28) % POWER 0 0 0 (29) OTHER STATUS (30) NA METHOD OF DISCOVERY B (31) DISCOVERY DESCRIPTION (32) Local Leak Rate Test Program

16 ACTIVITY RELEASED Z (33) CONTENT OF RELEASE Z (34) AMOUNT OF ACTIVITY (35) NA LOCATION OF RELEASE (36) NA

PERSONNEL EXPOSURES
17 NUMBER 0 0 0 (37) TYPE Z (38) DESCRIPTION (39) NA

PERSONNEL INJURIES
18 NUMBER 0 0 0 (40) DESCRIPTION (41) NA

LOSS OF OR DAMAGE TO FACILITY (43)
19 TYPE Z (42) DESCRIPTION NA

PUBLICITY
20 ISSUED N (44) DESCRIPTION (45) NA

NRC USE ONLY

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January 22, 1985

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Dear Sir:

Subject: Oyster Creek Nuclear Generating Station
Docket No. 50-219
Licensee Event Report Update

This letter forwards Reportable Occurrence No. 50-219/83-09/01X-2, a Licensee Event Report (LER) revision, in compliance with paragraph 6.9.2.a.2 of the Technical Specifications. The previous revision of this LER contained incorrect leak rates for three (3) valves (V-27-1, 3 and 4). This occurred due to an administrative error resulting from a procedure revision which had changed the leak rate data sheet number for the above valves.

Very truly yours,

PB Fiedler

Peter B. Fiedler
Vice President and Director
Oyster Creek

PBF:PC:dam
Enclosures

cc: Dr. Thomas E. Murley, Administrator
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NRC Resident Inspector
Oyster Creek Nuclear Generating Station
Forked River, NJ 08731

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