NRC Form 366
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/88

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Oyster Creek, Unit 1	0 5 0 0 0 2 1 9	1 OF 0 4

TITLE (4) REACTOR SAFETY VALVE LEAKED DURING HYDRO TEST DUE TO MANUAL LIFTING NUT COTTER PIN FAILURE

EVENT DATE (5)			LER NUMBER (6)					REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR		SEQUENTIAL NUMBER		REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBER(S)
												0 5 0 0 0
1 1	1 1	8 7	8 7	-	0 4 6	-	0 0	0 1	0 7	8 8		0 5 0 0 0

OPERATING MODE (9)	N	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)			
POWER LEVEL (10)	0 0 0	20.402(b)	20.405(c)	50.73(a)(2)(iv)	73.71(b)
		20.405(a)(1)(i)	50.36(c)(1)	50.73(a)(2)(v)	73.71(c)
		20.405(a)(1)(ii)	50.36(c)(2)	50.73(a)(2)(vii)	X OTHER (Specify in Abstract below and in Text, NRC Form 366A)
		20.405(a)(1)(iii)	50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	
		20.405(a)(1)(iv)	50.73(a)(2)(ii)	50.73(a)(2)(viii)(B)	VOLUNTARY
		20.405(a)(1)(v)	50.73(a)(2)(iii)	50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER
John P. McGrane, Operations Engineer	AREA CODE 6 0 9 9 7 1 - 2 4 2 5

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	
X	J C	R V	D 2 4 3	Y							

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE)	X NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

On November 10, 1987 at approximately 1709 hours, safety valve V-1-162 "F" was found to be leaking during a reactor vessel hydrostatic pressure test. At the time of this event, the reactor was shutdown at approximately 200 psig and a reactor vessel hydrostatic pressure test was in progress. Maintenance personnel investigated the cause of the safety valve leaking and discovered a cotter pin designed to hold the valve manual lift nut in place had failed. The nut had rotated down the valve stem threads to a position which induced a preload tension when the valve cooled opposing the force of the closing spring. This would allow valve seat leakage at a reduced pressure but would not have affected the actual valve lift setpoint when the valve cooled down. The nut was removed and the vessel hydrostatic pressure test was successfully completed. Afterward, the safety valve was replaced and the manual lift nuts were removed from all installed reactor safety valves to prevent this event from recurring. The safety significance of this event was minimal as neither nuclear nor personnel safety was affected.

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TEXT (If more space is required, use additional NRC Form 366A's) (17)

Date of Occurrence

This event occurred November 10, 1987 at approximately 1709 hours.

Identification of Occurrence

A reactor safety valve (IEEE Code RV) was found leaking with reactor pressure at approximately 200 psig during a reactor vessel hydrostatic pressure test.

This is a voluntary report.

Conditions Prior to Occurrence

The reactor mode switch was locked in the SHUTDOWN position, reactor temperature was 198°F and pressure was approximately 200 psig. A reactor vessel hydrostatic pressure test was in progress.

Description of Occurrence

On November 10, 1987 at approximately 1709 hours safety valve V-1-162 "F" was found to be leaking during a reactor vessel hydrostatic pressure test. The test was immediately discontinued and pressure was relieved from the reactor vessel. When the safety valve was inspected the cotter pin used to lock the manual lifting nut in place was found to be broken. The nut had rotated into a position which interfered with the valve closing spring tension. The manual lift nut on the affected valve was removed and the hydrostatic pressure test was successfully completed. Following the hydrostatic pressure test, the affected safety valve was replaced with a bench tested safety valve. Additionally, the manual lift nuts were removed from all the installed reactor safety valves to prevent this event from recurring.

Apparent Cause of Occurrence

The cause of this event was the mechanical fatigue failure of the manual lift nut cotter pin and subsequent manual lift nut repositioning which restricted valve stem thermal contraction, thus decreasing the closing spring tension. Vibration and thermal cycling were the mechanisms that caused the cotter pin failure and nut movement. (See attached drawing for details.) This resulted in the valve disk

position at pressures greater than 200 psig being slightly off the seat causing seat leakage. As the safety valve would start to lift in this condition, the stem thermal contraction preload force on the closing spring tension would be relieved within mils of stem movement.

Analysis of Occurrence and Safety Assessment

The purpose of the reactor safety valves is to prevent overpressurization of the reactor vessel should a transient occur where the normal reactor pressure control mechanisms are not adequate and reactor pressure exceeds 1212 psig. The safety valves discharge directly to the drywell atmosphere. The safety valve that leaked at approximately 200 psig (set for 1230 psig) during the hydrostatic pressure test, discharged reactor water into the drywell.

This event presented two safety concerns:

1. Unidentified primary coolant leakage increased temporarily.

2. Personnel in the drywell might have been contaminated from the reactor water discharging from the safety valve.

These safety concerns are not considered significant because:

1. The design and location of the safety valves precludes the possibility of the reactor core becoming uncovered when a safety valve leaks with plant conditions as they existed during this event. The leakage was very low and the capacity of the make-up water systems would have maintained reactor water level.

2. The radioactivity level of the reactor water at the time of this event was sufficiently low so as not to pose a hazardous contamination concern to personnel in the drywell.

Had this event occurred during a reactor start-up, the affected safety valve may have leaked and required a controlled plant shutdown.

Personnel safety could have been affected with the reactor operating at an elevated pressure had personnel been in the drywell when the safety valve leaked. However, a release to the environment would not have occurred, as the drywell is designed to withstand the level of energy release resulting from a leaking safety valve at all normal operating power levels. The replacement safety valve was bench tested and found to lift at the proper pressure (1230 psig) and to have zero seat leakage prior to lifting. All the installed safety valves performed as designed during the second hydrostatic pressure test. The actual setpoint of the valve and its ability to perform its intended function were not affected.

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TEXT (If more space is required, use additional NRC Form 366A's) (17)

Corrective Action

The valve that leaked at approximately 200 psig during the reactor hydrostatic pressure test was replaced with a bench tested safety valve. The manual lift nuts were removed from all other installed reactor safety valves to prevent this event from recurring. The maintenance procedure for the installation of reactor safety valves will be revised to include steps requiring the manual lift nut be verified removed prior to installation.

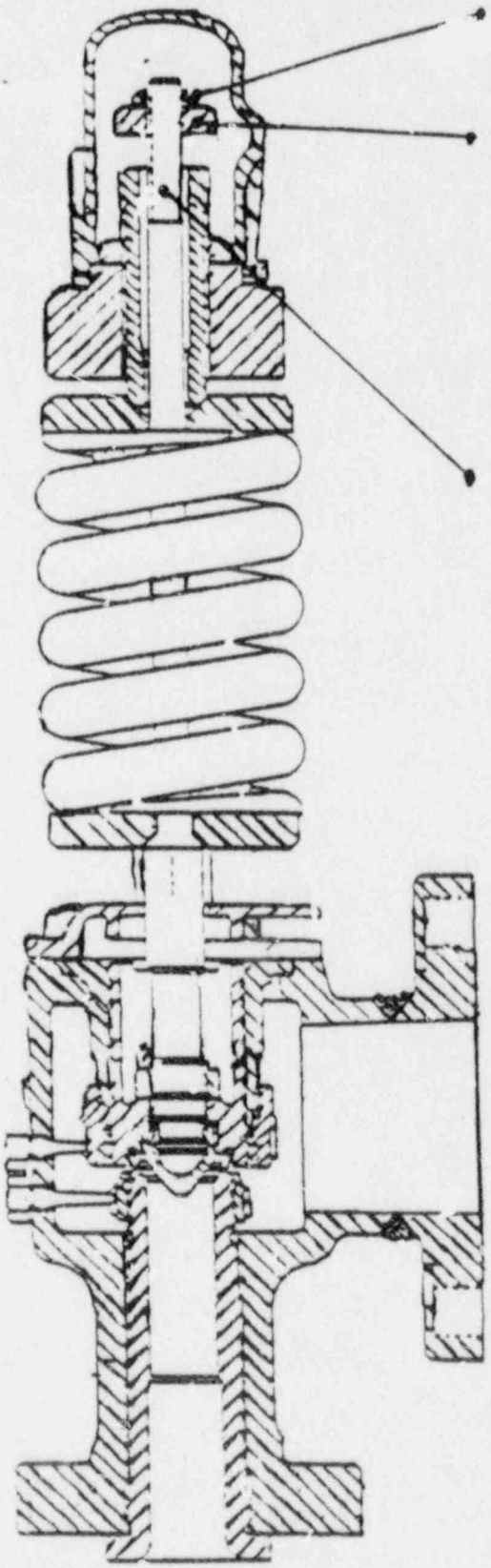
Similar Events

None

Failure Data

1. Manufacturer: Dresser
2. Model No.: 3777QA
3. Set Point: 1230 psig

0374A



Cotter Pin (which failed)

Manual Lift Nut (which rotated down, causing a valve opening preload tension when the valve cooled.

Threaded valve stem

TYPICAL
SAFETY
VALVE

GPU Nuclear

GPU Nuclear Corporation
Post Office Box 388
Route 9 South
Forked River, New Jersey 08731-0388
609 971-4000
Writer's Direct Dial Number:

January 7, 1988

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

Dear Sir:

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

This letter forwards one (1) copy of Voluntary Report No. 87-046.

Very truly yours,

Peter B. Fiedler
Vice President and Director
Oyster Creek

PBF:JR:dmd (#0374A)
Encs.

cc: Mr. William T. Russell, Administrator
Region I
U.S. Nuclear Regulatory Commission
631 Park Avenue
King of Prussia, PA 19406

NRC Resident Inspectors
Oyster Creek Nuclear Generating Station

Mr. Alex Dromerick
U.S. Nuclear Regulatory Commission
7920 Norfolk Avenue, Phillips Bldg.
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