NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/85

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): INADEQUATE SUPPORT OF REACTOR SAMPLE LINE DUE TO MISCOMMUNICATION OF TECHNICAL DATA CAUSED BY PERSONNEL ERROR

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 4	1 2	8 8	8 8	— 0 0 7 —	0 0	0 5	1 1	8 8		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9):

POWER LEVEL (10): 1 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)

[ ] 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)	[ ] 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)	[X] 50.73(a)(2)(ii)	[ ] 50.73(a)(2)(viii)(B)	
[ ] 20.405(a)(1)(v)	[ ] 50.73(a)(2)(iii)	[ ] 50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (AREA CODE)
John Galanto, Plant Engineering	609 971-4349

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

SUPPLEMENTAL REPORT EXPECTED (14)

[ ] YES (If yes, complete EXPECTED SUBMISSION DATE) [X] NO

EXPECTED SUBMISSION DATE (15): MONTH | DAY | YEAR

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

On April 12, 1988 it was determined that a Reactor Coolant sample line containing a containment isolation valve was not adequately supported. The discovery was made several days earlier by contractor personnel but a preliminary evaluation showed no problem existed. Later, when additional information was obtained it was determined that the valve weight was greater than the limit established by the analysis. The event is attributed to personnel error because in 1986 the valve in the line was replaced with a valve approximately three (3) times heavier than the original valve. When the weight increase was analyzed the information was miscommunicated on the actual orientation of the valve so the analysis was not accurate. The safety significance of this occurrence is considered minimal because present evaluations have determined that no piping failures would have occurred although code allowable stresses would have been exceeded in a design basis seismic event. Additionally, the inside containment isolation valve would have remained operable and would have isolated the line. The line will remain isolated until additional supports are installed. This LER will be distributed as required reading for plant engineering personnel.

IE22

NRC Form 366 (9-83)

DATE OF OCCURRENCE

This event was identified as reportable on April 12, 1988.

IDENTIFICATION OF OCCURRENCE

A reactor coolant sample line was not adequately supported after a valve was replaced. This condition placed the plant in an unanalyzed condition and is reportable under 10CFR50.73(a)(2)(ii)(B).

CONDITIONS PRIOR TO OCCURRENCE

The plant was in all modes of operation since October of 1985 which was the last time the line was known to be adequately seismically supported.

DESCRIPTION OF OCCURRENCE

On April 6, 1988 a walkdown of existing plant configuration was performed by contractor personnel during preparation for work in the area of reactor coolant sampling piping. It was noted during the walkdown that a valve was supported by nylon rope tied to its actuator. This finding was reported to a contractor site representative to determine if the valve was containment isolation valve, V-24-30. On April 8, 1988 it was determined that the valve being supported by the rope was indeed V-24-30. The operations department was notified and as a precaution isolated the line. The operations manager requested engineering to determine if a problem existed with the current configuration of supports. A preliminary engineering evaluation performed on April 8th determined that if the rope was removed no problem existed with piping stresses. At this time, it was recommended by engineering to leave the rope in place until the evaluation was finalized. The sample line was considered operable and returned to service. However, additional information became available on Tuesday, April 12th that indicated a problem did exist. The valve (V-24-30) was replaced in 1986 with a valve weighing approximately twenty-nine (29) pounds. The original valve weighed approximately ten (10) pounds. The evaluation performed on April 8th had concluded that a valve weight of up to 18 pounds was acceptable. On April 12, 1988 the operations department was informed that the current configuration did not meet ANSI B31.1 piping code allowable stresses for seismic qualification. Operations personnel declared the valve inoperable and isolated the line as required by Technical Specifications, using a primary containment isolation valve in series with the affected valve. On April 13, 1988, it was identified that a pipe support just upstream of the valve was broken. However, this hanger was not considered for support in the engineering evaluation performed on April 8, 1988.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Oyster Creek, Unit 1	0 5 0 0 0 2 1 9	8 8	— 0 0 7	— 0 0	0 3 OF 0 4

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

APPARENT CAUSE OF OCCURRENCE

During maintenance activities to fix a leaking valve (V-24-30) in 1986, it was discovered that the valve was obsolete and replacement parts did not exist. It was decided to replace the existing valve with a valve available from warehouse stock. Maintenance personnel requested engineering to evaluate the stock valve as a suitable replacement and to consider the additional weight of the new valve (almost three times heavier) for possible installation of additional supports. Plant Engineering developed a sketch and filled out a data sheet to show the valve configuration. This information was then used by an engineering mechanics engineer offsite who determined no additional supports were required. The new valve was installed based on this information. At some point, either during installation or thereafter the rod hanger supporting the piping upstream of the valve broke or was broken. Additionally, due to the horizontal mounting of the valve, personnel apparently tied a rope around the valve operator for temporary support during installation although this is not positively known. It is also not known how long the pipe hanger was broken but a picture of the area taken in October of 1985 clearly shows the hanger intact. The cause of this event is attributed to a miscommunication of technical information with respect to the orientation of the valve and its operator. The inaccurate information utilized by the offsite engineer resulted in an inaccurate analysis which indicated the line met seismic design criteria.

ANALYSIS OF OCCURRENCE AND SAFETY ASSESSMENT

The safety significance of this occurrence is considered minimal because analysis shows that although the piping configuration did not meet the maximum allowable stresses required by ANSI B31.1 the piping would have remained operable when supported by either the hanger or the rope during a seismic event. The inside containment isolation valve (V-24-29) in the sample line would have remained fully operable. The other containment isolation valve (V-24-30) near the broken hanger may have lost its electrical power to its solenoid operator but would have failed safe in the closed position.

CORRECTIVE ACTION

The sample line was declared inoperable and isolated. Additional supports have been designed and are scheduled to be installed. The line will remain isolated until the supports are installed. This LER will be distributed as required reading to all plant engineering personnel to re-enforce the importance of effectively communicating technical data.

NRC Form 366A
(9-83)

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/85

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Oyster Creek, Unit 1	0 5 0 0 0 2 1 9	8 8	— 0 0 7	— 0 0	0 4 OF 0 4

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

SIMILAR EVENTS

86-022: Control Rod Drive Hydraulic Control Units not Installed Per Design
86-021: Plant Systems Did Not Meet Seismic Design Basis

0492A

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:

May 11, 1988

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

This letter forwards one (1) copy of Licensee Event Report (LER) No. 88-007.

Very truly yours,

E E Fitzpatrick

E. E. Fitzpatrick
Vice President & Director
Oyster Creek

EEF:KB:dmd(0492A)
Enclosures

cc: Mr. William T. Russell, Administrator
Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Mr. Alexander W. Dromerick
U.S. Nuclear Regulatory Commission
Washington, DC 20555

NRC Resident Inspector
Oyster Creek Nuclear Generating Station
Forked River, NJ 08731

IE22
1/1