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 3

TITLE (4) CONTAINMENT PARTICULATE MONITOR SAMPLE LINE ISOLATION VALVES - POTENTIAL FAILURE TO OPERATE DUE TO DESIGN DEFICIENCY

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	2 0	8 8	8 8	— 0 0 3 —	0 0	0 6	1 6	8 8		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9)	N
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)
Roger B. Gayley, Operations Engineer	6 0 9 9 7 1 - 4 9 6 1

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

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS

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)

The plant's Containment Particulate Monitor sample line isolation valves' control circuitry does not meet single failure criteria as required by 10CFR50, Appendix A, General Design Criterion. This condition was determined reportable on May 20, 1988, while the plant was operating at rated output. The apparent cause of the condition is a design deficiency which has been present since the Containment Particulate Monitor was installed in 1976. This condition is significant in that it could place the plant outside its design basis containment leak rate during a design basis accident. A conservative analysis indicates that under the design basis loss of coolant accident with this single failure present, operators would have to diagnose and take corrective action within 6.7 hours to prevent exceeding 10% of the 10CFR100 limits. Direction by means of a standing order has been provided to operators to deenergize the power supply to close these valves upon a reactor low-low water level or a high drywell pressure alarm. Corrective action will be taken in accordance with the Integrated Living Schedule for the Oyster Creek plant.

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NRC Form 366A

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LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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

Date of Discovery

The condition was determined to be reportable on May 20, 1988. It has, however, existed since 1976 when the Containment Particulate Monitor was installed.

Identification of Occurrence

The control circuitry for Containment Particulate Monitor sample line isolation valves (V-38-9, V-38-10, V-38-16 and V-38-17) (EIIS Code IP) does not meeting single failure criteria as required by Nuclear Regulatory Commission General Design Criteria 21, 24, 54, and 56. This condition is considered reportable in accordance with 10CFR50.73(a)(2)(ii)(B).

Conditions Prior to Occurrence

The plant was operating at rated thermal output generating approximately 650 MWe when the condition was determined to be reportable.

Description of Occurrence

A sample line connected to the drywell air space is connected to the plant's Containment Particulate Monitor (CPM) equipment. Isolation is provided by two isolation valves in series located in the CPM inlet sample lines (V-38-9, V-38-10) and outlet sample lines (V-38-16 and V-38-17). These valves are automatically closed upon receipt of a containment isolation signal. The existing control circuitry configuration allows a single failure to occur (i.e., a short circuit between the power feed and control wires) that could render the valves remotely inoperable and stuck in the open position.

This condition was first identified during January 1988 during engineering of a CPM modification. The engineer who identified the condition initiated a "Preliminary Safety Concern and Potential Licensee Event Evaluation" (PSC). A Licensing Action Item (LAI) was initiated as a result of the PSC. A preliminary determination was made on the LAI by the end of April 1988. A final determination on reportability was made on May 20, 1988.

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

Apparent Cause of Occurrence

The apparent cause of this occurrence is a design deficiency. Section 6.2.4.5 of the Oyster Creek Final Safety Analysis Report (FSAR) under "Design Evaluation of Isolation Valves" indicates that isolation valves and their associated electrical systems are designed to prevent common mode failures. The existing configuration could allow a short circuit between the power feed and control wires that would render all four isolation valves inoperable and stuck in the open position. The existing configuration is therefore not in compliance with the original plant design criteria.

Analysis of Occurrence and Safety Assessment

Containment isolation is provided to contain the reactor coolant thermal energy and fission products which could be rele d during a Loss of Coolant Accident (LOCA). The Primary Containment desig sis leak rate could be exceeded if the unlikely events listed below were to occu simultaneously:

1. A single failure as designed above

2. A LOCA

3. Significant fuel failure as a result of the LOCA

A conservative analysis indicates that under the design basis loss of coolant accident with this single failure present, operators would have to diagnose and take corrective action within 6.7 hours to prevent exceeding 10% of the 10CFR100 limits.

Corrective Action

Direction by means of a standing order has been provided to operators to deenergize the power supply to these valves to close them upon receipt of a reactor low-low level or a high drywell pressure alarm. Corrective action will be taken to correct this deficiency in accordance with the Integrated Living Schedule (ILS).

Similar Occurrences

LER 87-040 Torus Oxygen Sample Line Does Not Meet Single Failure Criteria Due to Design Deficiency

(0510A)

GPU Nuclear

GPU Nuclear Corporation
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Forked River, New Jersey 08731-0388
609 971-4000
Writer's Direct Dial Number:

June 16, 1988

Director of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Mail Staton P1-137
Washington, DC 20555

Dear Sir:

Subject: Oyster Creek Nuclear Generating Station
Docket No. 50-219
Licensee ‑‑nt Report

This letter forwards one (1) copy of Licensee Event Report (LER) No. 88-003.

Very truly yours,

E E Fitzpatrick

E. E. Fitzpatrick
Vice President & Di.ecto-
Oyster Creek

EEF:GB:dmd(0510A)
Enclosures

cc: Mr. William T. Russell, Administrator
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