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): Oyster Creek, Unit 1

DOCKET NUMBER (2): 0 5 0 0 0 2 1 9

PAGE (3): 1 OF 0 4

TITLE (4): PARTIAL PRIMARY CONTAINMENT ISOLATION DURING TESTING DUE TO PROCEDURAL INADEQUACY

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	3 0	8 7	8 7	0 2 3	0 0	0 5	2 9	8 7		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): N

POWER LEVEL (10): 0 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)	[X] 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)	[ ] 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	
Robert J. Babecka, Operations Engineer	6 0 9	9 7 1 - 2 1 6 4

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)

[ ] 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 30, 1987, at 2310 hours while performing surveillance testing of reactor protection motor generator sets, a partial primary containment isolation occurred. The root cause of the event was procedural inadequacy. The surveillance procedure had not been revised to reflect two modifications to the plant, each of which would have caused a partial isolation independent of the other. In both cases the review for procedures that could be affected by the modification failed to identify this procedure as requiring changes. In at least one occurrence in November 1986, this test had been performed with a partial isolation occurring. It is possible that two other events occurred in October and November 1984. None of these were noted in the procedures or operator logs. The affected procedures and both modifications will be reviewed to ensure both modifications are reflected where necessary. Notification requirements have been emphasized to shift supervisory personnel. This report will be required reading for engineering and operations personnel. Guidance will be provided to improve engineering reviews of modifications.

IE22
1/1

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 7	— 0 2 3 —	0 0	0 2 OF 0 4

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

DATE OF OCCURRENCE

The event occurred on April 30, 1987 at 2310 hours.

IDENTIFICATION OF OCCURRENCE

During performance of surveillance testing, an unplanned partial primary containment (EIIS code JM) isolation occurred. This event is reportable in accordance with 10CFR50.73(a)(2)(iv).

CONDITIONS PRIOR TO OCCURRENCE

The reactor was in the SHUTDOWN mode. Reactor coolant temperature was less than 212°F. Procedure 619.2.019, "Reactor Protection M-G Set Generator Output Breaker Trip Test and Calibration", was being conducted.

DESCRIPTION OF OCCURRENCE

The prerequisites for surveillance testing of reactor protection system (RPS) motor generator set (MG) 1-1 (EIIS code JE, component MG), including installation of electrical jumpers, had been completed in accordance with procedure 619.2.019. The operators removed RPS MG 1-1 from service (i.e. the RPS transformer (EIIS code JE, component XFMR) 1-1 was to be used to supply power to RPS 1). The switching of power supplies caused the primary containment instrument air automatic isolation valve (V-6-395) (EIIS code LF, component ISV) and primary containment vent and purge isolation valves (EIIS code JM, component ISV) to close due to deenergization of RPS relays CR1 and 4K50 respectively. Recognizing the momentary deenergization of RPS 1, which occurred due to the power supply switch, had caused this partial isolation the operators reset the trip and opened the affected valves.

APPARENT CAUSE OF OCCURRENCE

The cause of the automatic closure of the primary containment vent and purge isolation valves was the momentary deenergization of a relay (CR1) (EIIS code JE, component RLY) when RPS MG 1-1 was secured. CR1 was added to the RPS as part of a modification which would cause isolation of primary containment ventilation if a high radiation condition existed in the primary containment. Deenergization of CR1 caused these ventilation valves to close. The root cause was procedural inadequacy. Procedure 619.2.019 did not mention that a primary containment ventilation isolation would occur or provide actions to prevent such an occurrence. Procedure 619.2.019 did contain a direction that when the RPS MGs were to be secured that procedure 408.2, "Removing the Reactor Protection System

MG Set 1-1 (1-2) from Service" was to be followed. Procedure 408.2 had been revised to include a precaution regarding this protective action, but the direction to follow procedure 408.2 was in the prerequisite section of 619.2.019 instead of the instruction section. Procedure 619.2.019 had been revised prior to the test to add statements to the instruction section which directed the operator to remove the RPS MGs from service but did not include a reference to procedure 408.2. Procedure 619.2.019 did not give adequate guidance to the operator to prevent this occurrence and was not one of the procedures reviewed for possible effects by the modification which installed CR1. During the investigation of this LER it was discovered that a similar ventilation isolation occurred during the previous performance of procedure 619.2.019 on November 14, 1986. The modification which installed CR1 was turned over on November 11, 1986. The revision to procedure 408.2 was not issued until March 1987. Procedure 408.2 was not one of the procedures reviewed for possible effects by the modification. It was revised after the modification was turned over when the effects of the modification on this procedure were independently identified. An interdisciplinary engineering review, which is frequently used to identify procedures that may require revision as a result of a modification, was not conducted for this modification. If this review had been conducted, both procedures may have been revised before the modification had been turned over. The event which occurred on November 14, 1986 was not logged nor reported.

The momentary deenergization of RPS 1 on April 30, 1987, also caused a second relay (4K50) (EIIS code JE, component RLY) to deenergize, which resulted in a reactor isolation signal. Closure of V-6-395 is included in this protective function. The remaining valves which could have been affected by a reactor isolation were previously closed. Relay 4K50 is energized when the intermediate range monitor control switch (EIIS code IB, component HS) is selected to ranges one through nine and is used to bypass the main steam low pressure signal when in SHUTDOWN, STARTUP or REFUEL modes.

The root cause was procedural inadequacy. Procedure 619.2.019 stated that performance of this test at system pressure less than 600 psig would result in a reactor isolation. A later step stated actions to prevent closure of V-6-395, however, the procedure had not been revised to reflect the modification which added 4K50. This modification was turned over in March 1984. An interdisciplinary engineering review had been conducted for this modification, but procedure 619.2.019 and 408.2 had not been recognized as being affected by the modification. Since this modification was installed, procedure 619.2.019 has been performed six times prior to this event. Of these performances three were with plant conditions which could have resulted in a reactor isolation signal. In addition to the occurrence in November 1986, the other possible occurrences were in October and November 1984. Review of the logs showed no mention of similar occurrences. Either the automatic closing function of V-6-395 was bypassed during these tests or its closure was not logged.

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 7	0 2 3	0 0	0 4 OF 0 4

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

ANALYSIS OF OCCURRENCE

The containment inerting system (EIIS code VB) is used to perform the functions of purging and makeup of the primary containment using either nitrogen or air. In the event of indications of a loss of coolant accident or a high radiation condition, the RPS isolates the vent and purge lines to minimize the release of containment atmosphere to the environment. A reactor isolation occurs upon indications of a steam line break outside of primary containment, a loss of coolant accident, or a gross fuel clad failure. This surveillance has been performed at power and, due to the configuration of the RPS logic, no protective actions occurred. This event resulted only in an unintentional partial primary containment isolation, no required protective functions were diminished, therefore the safety significance is minimal.

CORRECTIVE ACTION

The short term corrective action will be to review and revise procedures 619.2.019 and 408.2 to resolve problems noted. Both modifications discussed will have interdisciplinary engineering reviews conducted to identify any other procedures that may be affected. Notification requirements have been re-emphasized to shift supervisory personnel. This report will be made required reading for engineering and operations personnel. The long term solution will be to improve engineering reviews for modifications by providing guidance to plant contacts in the performance of those reviews. Existing procedures governing modification control will be reviewed to determine the appropriate vehicle for such guidance.

SIMILAR OCCURRENCE

LER 86-026: Reactor Scram During Excess Flow Check Valve Testing

(0338A)

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 29, 1987

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. 87-023.

Very truly yours,

Peter B. Fiedler
Vice President and Director
Oyster Creek

PBF:MH:dmd(0338A)
Enclosures

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

Mr. Alexander W. Dromerick
U.S. Nuclear Regulatory Commission
7920 Norfolk Avenue, Phillips Bldg.
Bethesda, MD 20014
Mail Stop No. 316

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

IE22
1/1