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 1 9	1 OF 0 5

TITLE (4): Actuation of Engineered Safeguards Functions Due to Loss of "1D" Electrical Bus Caused by a Cable Fault.

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 0	0 2	8 8	8 8	0 2 2	0 0	1 0	3 1	8 8		0 5 0 0 0

OPERATING MODE (9)		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)
POWER LEVEL (10)		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
Roger B. Gayley, Operations Engineer	AREA CODE 6 0 9 9 7 1 - 4 4 6 1

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)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
X YES (If yes, complete EXPECTED SUBMISSION DATE)	NO		1 2	3 1	8 8

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

On October 2, 1988 at 1357 hours, power to 4160 volt bus 1D was lost when breaker 1D tripped and locked out due to a ground fault on the bus. The plant had been shutdown since September 30, 1988 for a refueling maintenance outage. One of the "B" phase leads was later found grounded at a point between the 1D bus and Emergency Diesel Generator No. 2 output breaker. The loss of the bus led to actuation of Engineered Safeguards Functions. At 0140 hours, on October 3, 1988, as part of the recovery effort following restoration of the 1D bus, power was momentarily lost to Vital AC Power Instrument Panel VACP-1 while restoring its normal power supply. This led to the actuation of some additional Engineered Safeguards Functions powered from VACP-1. The failed Diesel Generator cable will be analyzed to determine the cause of the cable fault. This event is reportable based on 10CFR50.73(a)(2)(iv).

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

DATE OF OCCURRENCE

This event occurred on October 2, 1988 at 1357 hours.

IDENTIFICATION OF OCCURRENCE

An electrical fault on an underground cable from the No. 2 Emergency Diesel Generator output breaker caused the loss of Electrical Bus 1D which led to the actuation of Engineered Safeguards Functions. A momentary loss of power to Vital AC Power Panel VACP-1 which occurred during the recovery effort following restoration of the 1D bus led to the actuation of additional Engineered Safeguards Functions. This event is reportable under 10CFR50.73(a)(2)(iv).

CONDITIONS PRIOR TO DISCOVERY

A refueling/maintenance outage was in progress. The Reactor mode switch was in Shutdown. The reactor coolant temperature was 149° F.

DESCRIPTION OF OCCURRENCE

On October 2, 1988 at 1357 hours, power to 4160 volt bus 1D (EIIS Code EB) was lost when breaker 1D tripped and locked out due to a ground fault on a cable (Component Code CBL5). The lock out of the faulted bus prevented the No. 2 Emergency Diesel Generator from starting. The loss of the bus caused Reactor Protection System Channel No. 2 to trip due to an interruption in it's power supply. A full reactor scram signal and a main steam line isolation occurred as designed because reactor pressure was less than 600 psig.

Standby Gas Treatment System 1 was manually started at 1415 hours due to the loss of normal reactor building ventilation. At 1506 hours, power was restored to Reactor Protection System Channel No. 2 and the reactor scram and main steam line isolation signal were reset. The cable fault was located and isolated at approximately 2130 hours and the 4160 volt 1D bus was returned to service at 2242 hours.

At 0140 hours on October 3, 1988, the Drywell and Torus vent and purge valves, the Drywell sump, and the Drywell Equipment Drain Tank isolated, and the circuitry for the Standby Gas Treatment System (already running in manual) realigned to auto start, while restoring the normal power supply to instrument panel VACP-1. Normal power was being restored as part of the recovery effort following restoration of bus 1D. All systems were returned to normal shortly after restoring the normal power supply to instrument panel VACP-1.

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APPARENT CAUSE OF OCCURRENCE

The trip of the 1D breaker was discovered to be caused by a ground fault on the cable between the 1D bus and the Diesel Generator No. 2 (DG #2) breaker. One of the B phase leads was discovered grounded at a point between the 1D bus and DG #2. The DG #2 cable is an extension of the 1D bus located in the turbine building as the DG #2 circuit breaker is located in the diesel generator building. The cable that failed will be analyzed to determine the fault cause. A supplement to this LER will report the cause of the cable fault.

ANALYSIS OF OCCURRENCE AND SAFETY ASSESSMENT

The loss of the 1D bus has been previously analyzed. Power was available to essential loads, because of the independence and redundancy of the AC power supplies.

The Engineered Safeguards Functions described above performed as designed. The reactor scram and main steam line isolation occurred because power was lost to Reactor Protection System (RPS) Channel No. 2. Because of the arrangement of the bypass contacts associated with the main steam line low pressure trip, a loss of electrical power to a single RPS bus results in a full scram and main steam line isolation if reactor pressure is less than 600 psig.

The isolation of the Drywell and Torus vent and purge valves, Drywell sump, the Drywell Equipment Drain Tank, and the realignment of the circuitry for the Standby Gas Treatment System to auto start occurred due to the momentary loss of power to instrument panel VACP-1. The momentary loss of power occurred while manually restoring the normal power supply to VACP-1, which provides power to Panel 11F and Panel 11R. Panel 11F controls the Drywell and Torus vent and purge valves, the Drywell sump isolation valves and the Drywell Equipment Drain Tank isolation valves. Panel 11R controls the Standby Gas Treatment System. When power was momentarily lost to these panels, relays de-energized which caused the actions to occur as described above. All systems were returned to normal shortly after restoring the normal power supply to VACP-1.

This event occurred while the plant was shutdown. Two out of three Shutdown Cooling pumps lost power, however, a single shutdown cooling pump was sufficient to remove decay heat. Even if the single shutdown cooling pump had failed, procedures and alternate methods were available that could have been utilized to remove decay heat. Had this event occurred while the plant was on-line, an automatic shutdown of the reactor would likely have occurred due to the loss of secondary plant systems.

ANALYSIS OF OCCURRENCE AND SAFETY ASSESSMENT Cont'd

Since power was available to essential loads, and the Engineered Safeguards Functions performed as expected, the safety significance of this event is considered minimal.

CORRECTIVE ACTIONS

Prior to Restart

1. The DG #2 cable that failed has been replaced.

2. The failed cable will be analyzed to determine the failure cause.

3. The cables for DG #2 that did not fail and the cables for DG #1 will be DC Hi-Pot tested at a minimum of 15 KV.

4. The 1B3 cable will be DC Hi-Pot tested at a minimum of 15 KV. This action will be taken due to failure of the 1A3 cable that occurred during July, 1988.

5. The need to test the underground cable to the dilution plant will be evaluated to determine its potential impact on the 1D bus if it should fail.

6 Ground grid system testing will be performed between the diesel building and the turbine building, including the ground connection for the Diesel Generator switchgear.

7. Off-site surge suppression will be evaluated and, if required, scheduled for installation.

8. Doble Testing will be done on DG #1 and DG #2 cables.

Long Term

1. Doble Testing and AC Hi-Pot testing will be evaluated as an alternative to DC Hi-Pot testing. The voltage level for DC Hi-Pot testing will also be evaluated .

2. A formalized PM program for electrical cables will be implemented prior to the next refueling outage.

3. A formalized ground test program will be developed.

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SIMILAR OCCURRENCES

LER 75-009, Electrical fault on 1C bus and subsequent trip of 1C breaker.

LER 77-004, Electrical fault on 1C bus and subsequent trip of 1C breaker.

(0605A)

GPU Nuclear

GPU Nuclear Corporation
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609 971-4000
Writer's Direct Dial Number:
October 31, 1988

Director of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Mail Station P1-137
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. 88-022.

Very truly yours,

E. E. Fitzpatrick
Vice President & Director
Oyster Creek

EEF:JR:smz(0705A)
Enclosures

cc: Mr. William T. Russell, Administrator
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Mr. Alexander W. Dromerick
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NRC Resident Inspector
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