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 5

TITLE (4): DESIGN DEFICIENCY IN CORE SPRAY PUMP LOGIC

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 1	2 9	8 5	8 5	0 0 3	0 0	0 2	2 7	8 5		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): N

POWER LEVEL (10): 0 9 5

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)	[X] 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)	[X] 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	
Paul F. Cervenka, Plant Engineering	6 0 9	9 7 1 - 4 8 9 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 January 29, 1985, a design deficiency was discovered in the Core Spray System booster pump failure logic. Discharge pressure of the booster pumps is utilized to detect a booster pump failure which will trip the failed pump and provide a start signal to the backup booster pump. Two events were identified which can cause this instrumentation to misinterpret Core Spray System status and result in the system not performing according to its original design intent. The cause of this occurrence is a deficiency in the original plant design.

Corrective action consisted of performing a modification to replace the pressure switches on the booster pump discharge with differential pressure switches. The differential pressure switches will sense differential pressure across the booster pump. This modification will allow the pump failure logic to perform as originally designed under all postulated conditions.

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DATE OF OCCURRENCE

January 29, 1985

IDENTIFICATION OF OCCURRENCE

The preliminary safety concerns were documented identifying two separate scenarios which could cause instrumentation in the Core Spray System to misinterpret Core Spray System status, and result in the system not performing according to its original design intent.

This occurrence is reportable as required by 10CFR50.73(a)(2)(i)(A) and 10CFR50.73(a)(2)(v).

CONDITIONS PRIOR TO OCCURRENCE

The reactor was operating in the "RUN" mode with a thermal power output of 1860 MW_{th}, generator load at 633 MW_e, and a reactor coolant temperature at 521^oF.

DESCRIPTION OF OCCURRENCE

The core spray, with the associated automatic depressurization, constitutes the low-pressure standby core cooling system, which provides an alternate supply of reactor cooling water after the improbable occurrence of a pipe break accident in the reactor primary system. The standby core cooling system thus prevents the fuel-cladding from melting by removing fission-product decay heat from the reactor following postulated accidents which result in loss of coolant and which would lower reactor water level to uncover the fuel.

The core spray system supplies cooling water after the reactor pressure is reduced to about 285 psig. This will supply cooling water before the reactor overheats with large-to-intermediate pipe breaks; however, to accommodate certain intermediate-to-small pipe breaks with the feedwater system not available, the automatic depressurization system will provide supplemental blowdown to reduce reactor pressure faster and thus permit core spray actuation before the fuel uncovers and overheats.

The core spray systems consists of two independent loops with each loop containing redundant main and booster pumps. Upon an automatic initiation signal, low low water level or high drywell pressure, the following sequence will occur:

1. The primary main core spray pump in each loop starts. If either main pump fails to start, or trips after starting, the associated backup pump will receive a start signal within ten seconds from the RV-29 pressure switches located on the discharge of the main pumps.

2. After receiving an automatic start signal and adequate discharge pressure at its associated main core spray pump, the primary booster pump in each loop will start. If the primary booster pump in either loop fails to start, or trips after starting, the associated backup booster pump will receive a start signal within five seconds from the RV-40 pressure switches located on the discharge of the booster pumps.

The RV-40 pressure switches are an integral part of both the Core Spray System and the Automatic Depressurization System. The original design intent of the RV-40 pressure switches is specified below:

1. Provide a trip signal to the primary core spray booster pump and initiate a start signal to the backup booster pump if the primary pump trips or fails to start within five seconds.

2. Provide one of three permissive signals for initiation of the Automatic Depressurization System.

The RV-40 pressure switch setpoint is 230 psig to actuate and 180 psig to reset.

The preliminary safety concerns identified two event scenarios which could cause the RV-40 pressure switches to misinterpret core spray system status, and result in the system not operating according to its original design intent. A description of each event is given below:

<u>EVENT #1</u>

Upon automatic initiation of core spray, the primary main and primary booster pumps start as designed satisfying the RV-40 pressure switches. A single failure of the primary booster pump is postulated to occur. The backup booster pump will not start until the pressure at the RV-40 switch drops below the reset point of 180 psig. The reactor pressure interlock, which prevents the parallel isolation valves from opening, will have the core spray system operating on minimum flow until the reactor pressure decreases to a value less than 285 psig. The main core spray pump alone, operating on minimum flow, may develop sufficient discharge pressure, as sensed at the RV-40 pressure switches, to prevent the reset point of 180 psig from being reached. This has the consequence of delaying the time to rated core spray flow until reactor pressure drops to less than 180 psig. When reactor pressure decreases to less than 180 psig, core spray flow will start and the pressure as sensed at the RV-40 switch will fall below the reset point. The backup booster pump will now start and together with the main core spray pump will deliver rated core spray flow to the reactor vessel.

Analysis and calculations have determined that Event #1 is possible, and actual data collected under the postulated conditions is supportive of this analysis.

EVENT #2

Upon automatic initiation of Core Spray, the primary main and primary booster pumps start as designed satisfying the RV-40 pressure switches. For a large break design basis accident, reactor vessel pressure will decrease rapidly. As reactor pressure decreases, core spray system flow will increase and the booster pump discharge pressure, as sensed at the RV-40 pressure switch, will decrease. It is postulated that at very low reactor pressures (64 psig for Core Spray System I and 23 psig for Core Spray System II) the booster pump discharge pressure will fall below the reset value of 180 psig of the RV-40 pressure switches. This will result in the tripping of the primary booster pumps and starting of the backup booster pump. Combining this with a single failure of the backup booster pump results in the associated core spray system operating with only a main pump running.

Analysis and calculations have determined that Event #2 is an expected occurrence under the postulated conditions, however no tests were performed to confirm this.

APPARENT CAUSE OF OCCURRENCE

The cause of this occurrence is a deficiency in the original plant design. The use of pressure switches in the booster pump failure logic may prevent the system from functioning as designed under all postulated conditions.

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 5	0 0 3	0 0	0 5 OF 0 5

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

ANALYSIS OF OCCURRENCE AND SAFETY ASSESSMENT

The safety significance associated with each of the events described above is a loss of system redundancy. The safety analysis for Oyster Creek assumes two operable Core Spray Loops each of which is single failure proof. In each event, a single failure will result in only one Core Spray Loop being fully operable. Since our current Appendix K ECCS analysis assumes two loop operability, failure to inject with two loops for certain postulated LOCA events puts the plant in a condition not specifically analyzed in accordance with Appendix K criteria.

CORRECTIVE ACTION

The corrective action consisted of placing the reactor in a cold shutdown condition on February 2, 1985 to perform a modification to the Core Spray System.

The modification consists of replacing the existing RV-40 pressure switches located at the booster pump discharge with differential pressure switches across the booster pump suction and discharge lines.

The installation of the differential pressure switches will be completed prior to startup. The differential pressure switch modification will allow the core spray system to operate according to its original design intent without being susceptible to the events described within this report. All necessary procedures will be revised and operator training will be accomplished by means of required reading of a modification summary sheet.

FAILURE DATA

None

GPU Nuclear

GPU Nuclear Corporation
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609 971-4000
Writer's Direct Dial Number:

February 27, 1985

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. 85-003.

Very truly yours,

Peter B. Fiedler
Vice President and Director
Oyster Creek

PBF:KB:dam (#0472A)
Enclosures

cc: Dr. Thomas E. Murley, Administrator
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NRC Resident Inspector
Oyster Creek Nuclear Generating Station
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