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)

EMERGENCY SERVICE WATER SYSTEM SEISMIC CONCERNS

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 2	1 1	8 5	8 5	0 2 3	0 0	0 1	2 7	8 6		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)					
POWER LEVEL (10)	0 0 0	20.402(b)		20.405(c)		50.73(a)(2)(iv)	73.71(b)
		20.405(a)(1)(i)		50.36(c)(1)	XX	50.73(a)(2)(v)	73.71(c)
		20.405(a)(1)(ii)		50.36(c)(2)		50.73(a)(2)(vii)	XX 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)	XX	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)	50.73(a)(2)(vi)

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER
	AREA CODE
MARK SANFORD, Manager, Mechanical Systems	2 1 0 1 1 2 1 9 1 - 1 2 1 1 1 7

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
YES (If yes, complete EXPECTED SUBMISSION DATE)	XX NO				

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

Abstract

Separate analyses of the Emergency Service Water (ESW) System piping revealed that:

1. Non-seismic system tie-ins and two containment spray (C.S.) heat exchanger relief valve lines could fail during a seismic event.

2. Deficiencies in the ESW piping supports could render the system inoperable during a design basis seismic event.

If the deficiency noted in the first analysis was not corrected, a scenario existed where the ESW System may not have provided adequate flow if pipe breaks occurred coincident with C.S. System demand. This condition has existed since original plant startup, and is attributed to original design error. Immediate corrective action was taken to raise the ESW Surveillance Test acceptance criteria to a level assuring adequate flow for all required plant condtions.

IE22
1/1

The deficiency noted in the second analysis was based upon an inspection of ESW piping during the 10M outage. The inspection showed that 20 pipe supports had excessive gaps, allowing pipe movements not assumed present during previous analyses. Subsequent analysis showed possibly excessive overloads on ESW piping during a design basis seismic event. Immediate corrective action was to repair the subject supports during the 10M outage to put the ESW piping in an operable stress condition during and after a seismic event.

Dates of Discovery

Condition 1, which describes non-seismic connections to the ESW System, was discovered and identified as reportable on December 11, 1985. Condition 2, which describes ESW pipe support deficiencies, was discovered and identified as reportable on December 13, 1985.

Identification of Discovery

Condition 1, in which non-seismic connections to the ESW System piping were identified, could have prevented the ESW System from providing required flow to the C.S. heat exchangers in the event of a loss of coolant accident (LOCA). This would result in a reduced capability for removal of heat energy from containment. This condition could have occurred during a seismic event in which the non-seismic piping is assumed to shear off coincident with C.S. demand. This condition is considered reportable under 10CFR50.73(a)(2)(v)(B) and (D) and (vi).

Condition 2, identifying deficiencies in ESW pipe supports, could have rendered the ESW System inoperable for a seismic event. Failure of the system during the event could result in the loss of access to the ultimate heat sink for some potential accident conditions, such as a large break LOCA. This condition is reportable under 10CFR 50.73 (a)(2)(ii)(B) and (a)(2)(v)(B) and (D) and (vi).

Conditions Prior to Discovery

Not applicable.

Description of Discovery

Condition 1:

The ESW System provides cooling water to the containment spray heat exchangers (ultimate heat sink). These heat exchangers remove heat generated in the primary containment during the design basis loss of coolant acccident (LOCA). There are two independent ESW Systems, each consisting of two full capacity pumps located at the intake structure and two heat exchangers located in the reactor building along with interconnecting piping. Each ESW System has 1-1/2" and

3" tie-ins from the chlorination system to a 14" ESW header underground, prior to the piping entering the reactor building.

During a review of the baseline engineering for a proposed chlorination system modification it was determined that the 1-1/2" and 3" chlorination lines could not be seismically qualified as the chlorination building in which the piping originates is itself a non-seismic structure. If the piping in the chlorination building were to break, the ESW backflow through the piping could not be isolated. Further analysis determined that the safety relief valves on two out of four C.S. heat exchangers were not adequately supported for a seismic occurrence. This would present a further source of system leakage following seismically induced pipe breaks.

It has been calculated that a minimum flow rate of 2370 gpm through an ESW loop is required to adequately remove decay heat from containment. An analysis was performed to determine the flow loss through the non-seismic pipe breaks. The results indicated that an ESW pump would be required to deliver a minimum of 2800 GPM in the test configuration in order to assure the required 2370 GPM for the C.S. heat exchangers in the assumed faulted mode. Prior to May 1981, flow rate data was not routinely obtained as permanent flow instrumentation was not installed. In Service Testing began in May of 1981 which required that flow rate be measured. A review of approximately 160 tests revealed only five instances in which the ESW pump delivered less than 2800 gpm.
the five readings ranged from 2600 to 2750 gpm.

Condition 2:

The NRC inspection in May 1985 and the GPUN Q/A 100% inspection of ESW supports showed 20 ESW seismic frame type pipe supports had excessive gaps. When the 1985 non-linear piping analysis was performed including the gaps in the supports, loads on adjacent supports showed excessive overloading. This caused the ESW piping to exceed operability stress limits.

Apparent Cause of Occurrence

The apparent cause of condition 1 was a design inadequacy in the tie-in piping between the chlorination system and the ESW system, and a design inadequacy in the supports for 2 heat exchanger's relief valves.

The apparent cause of condition 2 was a design inadequacy in the ESW pipe supports.

Analysis of Occurrence and Safety Assessment

Condition 1:

The effect of a reduced flow rate to the CS heat exchangers resulted in an unanalyzed condition. Assuming no actions were taken to correct the degradation in ESW flow, this event would ultimately result in the inability to operate Core Spray and Containment Spray pumps due to Net Positive Suction Head limits. It would be expected, however, that reasonable actions could have been taken to mitigate the consequences of this event (e.g. starting second ESW pump in the operating loop). Since the startup from the last refueling outage, Emergency Operating Procedures have been approved and issued which provide adequate direction to the control room operators to start the second ESW pump when Suppression Pool water temperature reaches 95 degrees F.

Due to the low probability of this event and the historically adequate (greater than 2800 gpm) ESW flow, the safety significance of this event is considered minimal.

Condition 2:

The deficiencies in the ESW system piping supports rendered the system inoperable for a seismic event, which is a condition outside the design basis of the Oyster Creek plant. This condition alone could have prevented fulfillment of the ESW system function of providing a heat sink for heat removal after a design basis seismic event.

Corrective Actions

Condition 1:

In order to assure that the Emergency Service Water System can fulfill its designed safety function, the procedure currently used to determine ESW pump operability has been revised to change the minimum acceptable flow rate from 2370 to 2800 gpm. In addition, an engineering evaluation will be performed to determine if a system modification is required.

Condition 2:

The 20 subject supports were repaired to reduce the excessive gaps to acceptable design limits. One additional support was also strengthened. This put the ESW piping into an operable stress condition during and after a seismic event.

(0136A)

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:

January 27, 1986

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. 85-023.

Very truly yours,

Peter B. Fiedler
Vice President and Director
Oyster Creek

PBF:JR:dam(0136A)
Enclosures

cc: Dr. Thomas E. Murley, Administrator
Region I
U.S. Nuclear Regulatory Commission
631 Park Avenue
King of Prussia, PA 19406

Mr. Jack N. Donohew, Jr.
U.S. Nuclear Regulatory Commission
7920 Norfolk Avenue, Phillips Bldg.
Bethesda, MD 20014
Mail Stop No. 314

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

IE22
1/1