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)
Limerick Generating Station - Unit 1	0 5 0 0 0 3 5 2	1 OF

TITLE (4)

An Engineered Safety Feature Actuation of Core Spray, PHR, HPCI & Diesel Generation

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 9	2 6	8 5	8 5	0 7 7	0 0	1 0	2 1	8 5		0 5 0 0 0
										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)	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
John C. Nagle, Senior Engineer, Licensing Section	AREA CODE 2 1 5 8 4 1 - 5 1 8 4

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
				N					

SUPPLEMENTAL REPORT EXPECTED (14)

YES (If yes, complete EXPECTED SUBMISSION DATE)	NO	EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR

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

Abstract: 85-077

On September 26, 1985, with the Unit at 41 percent power, while performing the High Pressure Coolant Injection (HPCI) System Steam Line Differential Pressure Timer Functional Surveillance Test, spurious Division IV trip signals were received. This Loss of Coolant Accident (LOCA) signal resulted in initiation signals to the D Core Spray pump, D Residual Heat Removal (RHR) pump, HPCI system, D-14 Diesel Generator and the B train of Control Room Emergency Fresh Air Supply System. In addition, load shedding in anticipation of diesel generator loading occurred and the following components were affected: D144 480 VAC load center, Reactor Enclosure Cooling Water (RECW), the B Control Rod Drive (CRD) pump, the D Control Room radiation monitor and the D Control Room Chlorine Analyser. This load shedding worked as designed and the affected systems were restored to operation. Despite the initiation signals to three Emergency Core Cooling Systems (ECCS), (Core Spray, RHR and HPCI) no injection occurred because the HPCI system was removed from service to perform the aforementioned surveillance test and RHR and Core Spray require a low reactor pressure permissive. The diesel generator started and came to speed; however, it did not load because the safeguard bus was not de-energized. The cause of the spurious trip signals was traced to a loose connection in the panel which supplies power to the respective trip units and transmitter for Division IV.

NRC Form 366 (9-83)

IE22 1/1

Description of the Event:

On September 26, 1985, with the Unit at 41 percent power, while performing the High Pressure Coolant Injection (HPCI) steam line differential pressure timer functional surveillance test, spurious Divsion IV trip signals were received. This Loss of Coolant Accident (LOCA) signal resulted in the Engineered Safety Feature (ESF) actuation of D Core Spray pump start, D RHR pump start, HPCI start signal and the D-14 Diesel Generator start. As designed the D-144 480VAC circuit breaker, which is fed from the 4160 volt safeguard bus, was automatically tripped. Other systems or components which also automatically tripped as designed included: the B Control Rod Drive Pump Reactor Enclosure Cooling Water, the D Control Room radiation monitor and the D Control Room Chlorine analyzer. The Reactor Enclosure Heating and Ventilation System isolated, as designed on a loss of power to the chlorine monitor.

The EIIS code for the effected system is JE.

Consequences of the Event:

The consequences of this event are minimal because all safety components performed as designed. The Core Spray and RHR pumps did not inject inventory into the vessel because reactor pressure was greater than the discharge pressure of either pump. Prior to the event, the HPCI outboard steam supply valve was closed to prevent an auto-start while the timer functional surveillance test was being performed. The diesel generator auto started and reached synchronous speed; however, the output breaker did not close because the safeguard 4160 VAC bus did not de-energize. In addition, the load shed, as a precautionary measure, performed as designed, and all affected systems were returned to service with no adverse consequences.

Cause of the Event:

This event was caused by a loose connection to the power supply of the fuse for the 24 volts DC Division 4 ECCS trip units and Rosemount transmitters. It is postulated that while attaching an alligator clip to a terminal strip in this panel during the surveillance test, the technicians inadvertently disturbed this loose connection. Independent post event testing was able to reproduce the event by disturbing the wire with the loose connection.

Corrective Actions:

The panel was inspected and the suspect connection was tightened. To minimize recurrance of this type of disturbance the surveillance test will be revised to eliminate the use of the Division IV trip unit power supply. The necessary trip signal will be provided by adjusting the trip unit rather than applying an external voltage signal.

Previous Similar Occurences

The following LERs discuss events which resulted in a ECCS initiations or signals and/or a Diesel Generator Start:

85-052
85-040
85-037

PHILADELPHIA ELECTRIC COMPANY

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October 21, 1985

Docket No. 50-352

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: Licensee Event Report
Limerick Generating Station - Unit 1

This LER deals with an Engineered Safety Feature actuation of the Core Spray, Residual Heat Removal, HPCI and Diesel Generators.

Reference:	Docket No. 50-352
Report Number:	85-077
Revision Number:	00
Event Date:	September 26, 1985
Report Date:	October 21, 1985
Facility:	Limerick Generating Station P.O. Box A, Sanatoga, PA 19464

This LER is being submitted pursuant to the requirements of 10 CFR 50.73(a)(2)(iv).

Very truly yours,

W. T. Ullrich
Superintendent
Nuclear Generation Division

cc: Dr. Thomas E. Murley, Administrator, Region I, USNRC
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See Service List

IE22
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Atomic Safety & Licensing Board Panel
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September, 1985