10 CFR 50.73

PHILADELPHIA ELECTRIC COMPANY

LIMERICK GENERATING STATION
P. O. BOX A
SANATOGA, PENNSYLVANIA 19464
(215) 327-1200 EXT. 2000

J. DOERING, JR.
PLANT MANAGER
LIMERICK GENERATING STATION

April 26, 1991

Docket Nos. 50-352
50-353
License Nos. NPF-39
NPF-85

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

SUBJECT: Licensee Event Report
Limerick Generating Station - Units 1 and 2

This LER reports the inadvertent start of a Unit 2 Emergency Diesel Generator (EDG) and the 'C' Emergency Service Water (ESW) system pump, both Engineered Safety Features, as a result of a spurious Loss of Coolant Accident signal. This event was due to a lack of attention to detail that resulted in an incomplete performance of a multiple part procedural step during surveillance testing of the EDG logic.

Reference:	Docket Nos. 50-352
	50-353
Report Number:	2-91-004
Revision Number:	00
Event Date:	March 27, 1991
Report Date:	April 26, 1991
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,

J. Doering

WGS:rgs

cc: T. T. Martin, Administrator, Region I, USNRC
T. J. Kenny, USNRC Senior Resident Inspector, LGS

IE22

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): Limerick Generating Station, Unit 2

DOCKET NUMBER (2): 0 5 0 0 0 3 5 3

PAGE (3): 1 OF 0 4

TITLE (4): The inadvertent start of an Emergency Diesel Generator and an Emergency Service Water pump due to a spurious LOCA signal as a result of personnel error.

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 3	2 7	9 1	9 1	— 0 0 4 —	0 0	0 4	2 6	9 1	Unit 1	0 5 0 0 0 3 5 2
										0 5 0 0 0

OPERATING MODE (9): 5

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: G. J. Madsen, regulatory Engineer, Limerick Generating Station

TELEPHONE NUMBER: AREA CODE 2 1 5 | 3 2 7 - 1 2 0 0

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)

[ ] 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 March 27, 1991, an inadvertent LOCA signal was generated while System Engineers (SEs) were performing the Division III Loss of Coolant Accident (LOCA) - Loss of Offsite Power (LOOP) logic system functional Surveillance Test (ST) procedure associated with the D23 Emergency Diesel Generator (EDG). This LOCA signal caused the Unit 2 D23 EDG to start, an Engineered Safety Feature (ESF) actuation. Additionally, the associated Division III AC Safeguard bus shed non-essential loads. After the D23 EDG start, the common 'C' Emergency Service Water (ESW) pump started automatically, also an ESF actuation. The actual consequences of this event were minimal in that no actual LOCA signal occurred and no adverse conditions resulted from the spurious signal. The cause of this event was personnel error in that the SE functioning as the test director in the Main Control Room (MCR) did not complete two substeps of a multiple part procedural step due to a lack of attention to detail. The SEs involved were counseled on the need for attention to details and the need to adequately communicate. A letter from the Limerick Generating Station Vice President to all site personnel was issued addressing this event and other incidents involving personnel error that have occurred during the current Unit 2 refueling outage. Additionally, the ST procedure and other similar ST procedures associated with the remaining Unit 1 and Unit 2 EDGs will be revised prior to their next performance to make the specific step into two steps.

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

Unit Conditions Prior to the Event:

Unit 1 Operational Condition was 1 (Power Operation) at 100% Power Level.

Unit 2 Operational Condition was 5 (Refueling) at 0% Power Level.

The D23 Emergency Diesel Generator (EDG) was administratively declared inoperable for the performance of the Division III Loss of Coolant Accident (LOCA) - Loss of Offsite Power (LOOP) logic system functional Surveillance Test (ST) procedure, ST-1-092-113-2, "D23 Diesel Generator 4 KV Safeguard Loss of Power LSF/SAA and Outage Testing." The Unit 2 'C' Residual Heat Removal (RHR, EIIS:BO) system pump and 'C' Core Spray System (CSS) pump were placed in operation in a full flow test configuration in accordance with the ST procedure. Additionally, the Unit 2 'B' RHR system was in service performing the function of shutdown cooling and was unaffected by this event.

Description of the Event:

On March 27, 1991, at 0520 hours, an inadvertent LOCA signal was generated while utility employed System Engineers (SEs) were performing the Unit 2 Division III LOCA-LOOP logic system functional test in accordance with ST procedure ST-1-092-113-2. This LOCA signal caused the Unit 2 D23 EDG (EIIS:EL) to start, an Engineered Safety Feature (ESF) actuation. Additionally, the associated Division III D23 4KV AC Safeguard bus automatically shed non-essential loads. After the D23 EDG start, the common 'C' Emergency Service Water (ESW, EIIS:BI) system pump (EIIS:P) started automatically, also an ESF actuation. Upon initiation of the D23 4KV Safeguard bus (EIIS:BU) load shed of non-essential equipment, the 'C' RHR system pump remained in operation (i.e., minimum flow line; no injection) while the full flow test valves for both the RHR system and CSS closed as designed. The 'C' CSS pump also tripped and then 10 seconds later actuated during load sequencing, as designed. However, the 'C' CSS pump did not inject into the Unit 2 reactor vessel due to the injection valve being blocked closed in accordance with the ST procedure. Additionally, the D23 EDG output breaker (EIIS:BKR) did not close and supply power to the D23 4KV Safeguard bus since this bus was being powered by the offsite power source at the time of the event. This is in accordance with the design of the EDG system. Immediate actions were taken by Main Control Room (MCR) operations personnel to verify that the Division III LOCA signal was spurious, to secure the D23 EDG and 'C' ESW system pump, and to reset the load shed, thereby, restoring Safeguard power to Division III non-essential loads.

A four hour notification was made to the NRC on March 27, 1991, at 0839 hours, in accordance with the requirements of 10CFR 50.72 (b)(2)(ii), since this event resulted in the inadvertent automatic actuation of ESFs. This report is being submitted in accordance with the requirements of 10CFR 50.73 (a)(2)(iv).

Analysis of the Event:

The actual consequences of this event were minimal in that an actual LOCA signal did not exist and no adverse conditions were created by the spurious LOCA signal

and equipment response. All of the equipment functioned as designed in response to the inadvertent LOCA signal. Additionally, the equipment associated with the load shed of the D23 Safeguard bus did not affect Unit 2 operations. Equipment required to be operable for Unit 2 was powered from the operable Division II and IV Safeguard buses, D22 and D24. Additionally, no Emergency Core Cooling System (ECCS) injections occurred nor were required. This testing is only performed while the reactor is shutdown and only on equipment that is not required to be operable, therefore, an inadvertent LOCA signal would not result during power or startup operations.

In the event an actual LOCA condition had occurred, the D22 and D24 EDGs, the Division II and IV Safeguard buses and associated loads were operable as required while shutdown and would have performed as designed.

Cause of the Event:

The primary cause of this event was personnel error in that the SE functioning as the test director in the MCR did not complete two substeps (i.e., substeps b and c) of procedural step 6.8.35 in procedure ST-1-092-113-2 due to a lack of attention to detail. Additionally, contributing causal factors are 1) less than adequate human factoring of procedural step 6.8.35, and 2) inadequate communication between the SE in the Auxilary Equipment Room (AER) and the SE in the MCR.

On March 27, 1991, three SEs were performing procedure ST-1-092-113-2 with two SEs located in the MCR, one as a test director and the other as a communicator. The third SE was located in the AER to perform the required test functions at this location. The SEs entered procedural step 6.8.35 which requires the three following actions:

a) at panel 20C640, place Emergency Core Cooling System modified test switch #1 to OFF,

b) at panel 20C601, depress Core Spray, Reset 3, (E21A-S17C) pushbutton, and

c) verify Reset 3 light is off.

These substeps are required to be performed to reset the CSS actuation logic signal which was initiated earlier during the performance of the procedure. Substep (a) is performed in the AER and substeps (b) and (c) are performed in the MCR. The SE in the AER performed substep (a) and informed the communicator in the MCR that the "step", meaning substep (a), was completed. The communicator then informed the test director in the MCR that the "step" was completed. The test director signed off procedural step 6.8.35 (i.e., substeps a, b, and c) as completed and continued on with procedure ST-1-092-113-2. The "Procedure" section of procedure ST-1-092-113-2 was then completed and the "Return to Normal" section was entered. Step 7.1.2 of procedure ST-1-092-113-2 requires that the Emergency Core Cooling System modified test switch #1, used in the procedure section, be removed from the AER panel 20C640. However, when the SE in the AER performed this step, a Division III LOCA signal was generated.

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

The D23 EDG started, the Division III 4KV AC Safeguard bus load shed and sequencing occurred, and the 'C' ESW system pump started.

Corrective Actions:

The SEs involved were counseled on the need for attention to detail and the importance of adequately communicating. A SE group all hands meeting was held on March 29, 1991, to review this event and reinforce management expectations for attention to detail and the importance of adequate communication. An evaluation was performed during the event investigation which determined that no previous similar occurrences existed involving SE personnel. Additionally, a letter from the Limerick Generating Station Vice President to all site personnel dated March 27, 1991, addressed this event and other incidents involving personnel error that have occurred during the current Unit 2 refueling outage. The letter re-emphasized management expectations regarding the need for attention to detail and the importance for all site personnel to fully understand the consequences of their actions before proceeding with their work.

Procedure ST-1-092-113-2 and seven other similar ST procedures, which are associated with the remaining Unit 1 and Unit 2 EDGs, will be revised, prior to their next performance, to make step 6.8.35 into two steps. One step (i.e., substep 6.8.35.a) will include the action performed in the AER and the second step (i.e., substep 6.8.35.b and c) will include the actions performed in the MCR.

Generic implications were considered by the plant staff and no negative trends could be identified at this time from the current data available. This event is included in the In-House Event and Investigation Program which enables plant management to track, trend, and analyze LGS events. This program is used for the identification of plant weaknesses and common mode failures. This program will aid in the identification of any future negative trends associated with this concern.

Previous Similar Occurrences:

LERs 1-84-024, 1-85-040, 1-86-025, 1-87-042, 1-88-010, 1-89-038, 2-90-013, and 1-91-005 also resulted in ESF actuations due to inadequate completion of a procedural step because of a lack of attention to detail. The corrective actions for the above listed LERs are considered adequate to mitigate this type of personnel error. This occurrence of personnel error within the SE group has been evaluated by the plant staff and at this time does not indicate an adverse trend based upon the In-House Event and Investigation Program. These types of personnel errors will continue to be monitored and if an adverse trend develops, appropriate corrective actions will be taken as deemed necessary.

Tracking Codes: (A2) - Failure to Follow Implementing Procedures
(A7) - Failure to Communicate
(D) - Procedure Deficiency