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 0 5 |

TITLE (4)
LOCA Signal Generated During Surveillance Test of Reactor Vessel Level Transmitter

| 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 | 2 5 | 8 7 | 8 7 | - 0 1 9 | - 0 0 | 0 6 | 2 4 | 8 7 | | 0 5 0 0 0 |
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| OPERATING MODE (9) | 5 | THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11) | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|
| POWER LEVEL (10) | 0 0 | | 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 |
|---|---|
| | AREA CODE |
| Charles A. Mengers, Senior Engineer, Licensing Section | 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 | |
|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | | | | | |
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SUPPLEMENTAL REPORT EXPECTED (14)

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

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


Abstract: 87-019

On May 25, 1987, at 0205 hours, an unplanned Loss of Coolant Accident signal resulted during a surveillance test on a Reactor Water Level transmitter. During performance of the surveillance test, the transmitter being tested was vented before being isolated. This caused pressure oscillations in the reference leg which affected other level transmitters, thereby, causing the actuation. The logic actuation resulted in a start of the 'C' Residual Heat Removal (RHR) and 'C' Core Spray pumps, a channel 'A' RPS scram signal, an ADS permissive signal and a D134 safeguard load center load shed. The actuation was confirmed, the logic was reset and the affected equipment was secured. The Low Pressure Coolant Injection (LPCI) mode of the (RHR) system did inject into the Reactor causing the water level in the Reactor Vessel and Reactor cavity to rise ten inches. After verifying that the Reactor Water level was normal, the RHR and Core Spray pumps were manually secured; by 0300 hours all of the equipment associated with the actuation was returned to normal operating conditions. There was no release of radioactive material and no adverse consequences associated with this event. This event was discussed on June 5 at the technician "All Hands Meeting" to emphasize the importance of resuming a procedure at the correct step.


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

Unit Conditions Prior to the Event:

Operating Mode 5 (Refueling)
Reactor Power-0%

Description of the Event:

On May 25, 1987 at 0205 hours, an unplanned Loss of Coolant Accident (LOCA) signal resulted during a surveillance test on Reactor Water Level transmitter, LT-42-IN181C. All engineered safety features (ESFs) associated with the transmitter being tested had been disabled in accordance with the procedure. Before beginning valving the transmitter out of service, the technician called Health Physics to survey the instrument rack; however, he resumed performance of the test procedure on the step where the transmitter is vented, instead of the previous step where the transmitter is isolated. While the technician was venting transmitter LT-42-IN181C, he observed water flowing into the head chamber above the valves he had opened. Realizing his error, he quickly closed the valves causing a pressure surge in the reference leg of the transmitter, which simulated a Reactor Vessel low water level signal in all of the transmitters sharing the reference leg.

In response to the low water level signals generated by the transmitters, the following systems responded as designed. The 'C' Core Spray pump started and the 'C' Residual Heat Removal (RHR) pump started as RHR aligned to the Low Pressure Coolant Injection (LPCI) mode of operation. The 'C' RHR pump injected water from the suppression pool into the Reactor Vessel, causing the water level to rise ten inches before it was manually secured. The 'C' Core Spray injection valve had previously been blocked closed to support outage activities, thus, preventing Core Spray from injecting. A channel "A" Reactor Protection System (RPS) half scram signal, an Automatic Depressurization (ADS) level 3 (Reactor Vessel Low Water Level) permissive signal and a D13 Diesel start signal were generated. The D13 Diesel had been blocked for a maintenance outage and did not start. Since either a coincident high drywell pressure signal and Reactor Vessel low water level signal or a 420 second time-delayed Reactor Vessel low water level signal is needed to satisfy the ADS logic, ADS did not actuate. Additionally, a D134 Safeguard Load Center load shed commenced with the LOCA signal, causing loss of power to the 'A' Reactor Water Cleanup (RWCU) pump, the 'A' instrument gas compressor, the 'C' Main Control Room chlorine

and high radiation detector, and the channel 'C' radiation detectors for the Reactor Enclosure and the Refueling Floor Area; resulting in isolations of the Main Control Room, the Reactor Enclosure and the Refueling Floor ventilation systems. The Control Room Emergency Fresh Air Supply (CREFAS) System and the Standby Gas Treatment System (SGTS), engineered safety features, responded to these isolations as designed.

The EIIS codes for the affected systems are CE, BO, JC, BM, VA, VI and DH for the RWCU, RHR, RPS, Core Spray, Reactor Enclosure ventilation, Main Control Room ventilation and SGTS, respectively.

Consequences of the Event:

There were no adverse consequences associated with this event. There was no release of radioactive material resulting from this event. The unit was shutdown for refueling and the reactor pressure vessel head had been removed prior to the event. The reactor vessel water level was at the flange. All Engineered Safety Features (ESFs), Emergency Core Cooling System (ECCS) and Reactor Protection System (RPS) actuations occurred as designed due to the Reactor Vessel Water low level signal. Because the D13 diesel generator and Core Spray injection valve were blocked for maintenance, consequences resulting from these initiation signals did not occur. This surveillance test is not normally performed during operating conditions.

A Health Physics technician was in the Reactor Vessel cavity when LPCI mode of RHR injected. Water level which was at the top of Reactor Vessel flange rose and began filling the Reactor Vessel Cavity. The technician exited the cavity and was not contaminated since he was wearing water tight protective clothing.

Cause of the Event:

Cause Code: Personnel Error (A)

The cause of the event was the vendor employed technician's failure to resume the surveillance test procedure at the correct step. During performance of the procedure, the technician stopped to call Health Physics to survey the instrument rack before he began valving. This break in continuity caused him to resume performance of the test on the step where the transmitter is vented instead of the previous step where the transmitter is isolated. When the technician opened the valves to vent the transmitter, he observed water flowing into the head chamber above the valves; realizing his error he quickly closed the valves. This action sent pressure oscillations through the reference leg simulating a Reactor Vessel low water level signal in all of the transmitters sharing this reference leg. The logic systems associated with each of the transmitters caused the resulting signals and actuations to occur.

Corrective Actions:

After verifying that a Reactor Vessel low level condition did not exist and that the actual water level was greater than 215 inches and rising, the Reactor Vessel low water level signal was reset. At 0206 hours, the 'C' LPCI injection and 'C' Core Spray RHR pumps were manually secured. Steps were immediately taken to reset the shunt trip controls associated with the D134 load shed. At 0228 hours the 'A' RWCU pump was started and the system was returned to service. At 0242 hours, the 'A' channel RPS half scram signal was reset. At 0246 hours, normal ventilation was restored to the main control room; and at 0250 hours, power was restored to the Reactor Enclosure fans and the Refuel Floor fans, establishing normal ventilation in these areas. Power was restored to the 'A' instrument air compressor at 0257 hours and at 0300 hours all shunt trip coils were verified to have been reset.

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

Actions to Prevent Recurrence:

The technician involved was counseled on the importance of resuming an unfinished surveillance test on the correct step. This event was discussed on June 5, 1987, at the technician "All Hands Meeting", emphasizing the importance of technicians assuring that prior to resuming a partially completed surveillance test, they know which steps have been completed.

Previous Similar Occurrence:

Limerick LERs 86-048, 85-040, and 85-037 reported ESF actuations resulting from a valving error during a surveillance test.

PHILADELPHIA ELECTRIC COMPANY

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P.O. BOX 8699

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June 24, 1987

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 concerns a Loss of Coolant Accident signal generated during a surveillance test of a Reactor Water Level transmitter while the reactor was shutdown for refueling.

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| Reference: | Docket No. 50-352 |
| Report Number: | 87-019 |
| Revision Number: | 00 |
| Event Date: | May 25, 1987 |
| Report Date: | June 24, 1987 |
| Facility: | Limerick Generating Station<br>P.O. Box A, Sanatoga, PA 19464 |

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

Very truly yours,

R. H. Logue
Assistant to the Manager
Nuclear Support Department

cc: W. T. Russell, Administrator, Region I, USNRC
E. M. Kelly, Senior Resident Site Inspector
See Service List

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