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 4

TITLE (4)
Actuation of Engineered Safety Features due to a 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 8	1 8	8 7	8 7	0 4 2	0 0	0 9	1 7	8 7		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 4
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	TELEPHONE NUMBER
Charles A. Mengers, 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

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)

Abstract:

On August 18, 1987 at 0341 hours a Division 4 Loss of Coolant Accident (LOCA) signal occurred. A spurious high Drywell pressure signal combined with an existing low reactor pressure resulted in the LOCA signal. The signal caused a number of automatic Engineered Safety Features to occur. All systems were verified to have functioned as designed and were returned to normal by 0432 hours. There were no adverse consequences and no release of radiation as a result of this event. The spurious high Drywell pressure signal was the result of a valving error during a surveillance test. A failure to follow approved procedures was the cause of the event. The vendor technicians involved were counseled regarding the importance of following procedures. In addition, this event was reviewed with Instrument and Control technicians at an "All Hands" meeting to emphasize the seriousness of the event.

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

Plant Operating Conditions Prior to the Event:

Cold Shutdown, operational condition 4

Reactor Power - 0%

Description of the Event:

On August 18, 1987 at 0341 hours a Division 4 Loss of Coolant Accident (LOCA) signal occurred when greater than 1.68 psig was sensed in a Drywell pressure sensing line. A valving error during the return to service section of ST-2-042-839-1, RPS and NSSSS-Drywell Pressure-High; DivisionIIB Channel B2/D Transmitter Response Time Test (PT-42-1N050D), caused the pressure perturbation. Various Engineered Safety Features (ESF) were initiated, in addition to control room annunciation. The affected pressure transmitters, PT-42-N094D and PT-42-N094H in conjunction with the existing low reactor pressure resulted in completion of the Division 4 LOCA logic. The following automatic actions occurred, as designed:

"D" Residual Heat Removal (RHR) pump start signal
"D" Low Pressure Coolant Injection (LPCI) injection valve opened.
High Pressure Coolant Injection initiation signal
"D" Core Spray (CS) pump start
"D" Low Pressure Coolant Injection (LPCI) injection valve opened.
"D-14" Diesel Generator (D/G) started and the associated load shed occurred.
"D" Emergency Service Water (ESW) pump start

The load shed caused:
1) a Group VIA Nuclear Steam Supply Shutoff System (NSSSS) isolation signal
2) a 'D' RHR Service Water (SW) radiation monitor to fail downscale, and
3) a control room ventilation isolation signal.

The effect of the automatic actions was minimal. Shutdown Cooling was in service at the time of the event on the 'B' loop with the 'D' RHR pump running. The net effect therefore was to inject return water (from the existing shutdown cooling suction) two ways, through the existing shutdown cooling return and also through the injection valve. Therefore, vessel make-up remained a closed loop and RPV water level did not change. All Group VIA NSSSS isolation valves (which include N_2 inerting

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supply and purge) were closed due to outage activities. Therefore, no valve movement occured. The RHR SW radiation monitor was "Bypassed" at the time of the event for maintenance work, and control room ventilation was isolated for testing. As a result, the automatic actions associated with each system were defeated and therefore had no impact.

At 0342 hours the 'D' LPCI injection valve was closed and the Core Spray pump secured. The "D"-14 D/G and ESW pump were removed from service at 0350. By 0432 hours all systems and electrical line-ups were normalized and the Group VIA isolation was reset in accordance with GP-8, Primary and Secondary Containment Isolation Verification and Reset.

The EIIS codes for the affected systems are: BO for LPCI, EK for the Emergency AC Supply System (Diesel Generators) JM for NSSSS, BJ for HPCI, BM for Core Spray, and IL for Radiation Monitoring.

Consequences of the Event:

There were no adverse consequences and no release of radiation as a result of this event. All affected systems were returned to normal line-up by 0432 hours.

If the event occurred during power operations, only a HPCI injection would occur. HPCI is initiated by greater than 1.68 psig Drywell pressure sensed by both of the affected instruments. Other ECCS systems are initiated by a LOCA signal which requires reactor pressure less than 455 psig and greater than 1.68 psig Drywell pressure. During power operations, reactor pressure is approximately 1000 psig which precludes an initiation by this part of the LOCA logic. The consequences of a HPCI injection for various reactor power conditions are mitigated by automatic level and pressure control systems and ESF actuations and do not present unanalyzed condition. Chapter 15.5.1 of the FSAR details an unplanned HPCI injection at full power. The worst case of this event would produce a high neutron flux scram. Other scenarios exist for unplanned coolant injections for non-power operation. If the reactor was in "Hot Shutdown" it is possible that a cold water injection could cause a cool down rate greater than that allowed by Technical Specifications. During Refueling activities, if the reactor cavity is flooded, an unplanned injection could cause an overflow condition of the cavity onto the refueling floor. An injection in excess of 10 minutes is necessary before the potential for an

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unplanned release would develop. It is expected that operator action would be sufficient to preclude this extreme.

Cause of the Event:

Cause code: Personnel Error A2

The cause of the event was cognitive personnel error. The vendor technicians performing the surveillance test failed to follow the approved procedure during the return to service section of the test. Consequently, steps were performed out of sequence.

Corrective Actions:

The valving error was immediately recognized, corrected, and the control room operator was informed of the mistake. Systems which initiated as a result of the error, were returned to normal by 0432 hours. The surveillance test was subsequently completed in accordance with procedure.

Action Taken to Prevent Recurrence:

The three individuals performing the test were counseled regarding the importance of following procedures. Also each received both written and oral reprimands emphasizing the seriousness of their mistake. In addition, this event was reviewed with Instruments and Control technicians at an "All Hands" meeting to emphasize the seriousness of the event.

Previous Similar Occurrences:

LERs 85-037, 85-040, 87-019 each detail ESF actuations which occurred as a result of instrument valving mistakes. Each event was caused by a personnel error.

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September 17, 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 details the unplanned actuation of Engineered Safety Features which occurred as a result of a personnel error.

Reference:	Docket No. 50-352
Report Number:	87-042
Revision Number:	00
Event Date:	August 18, 1987
Report Date:	September 17, 1987
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,

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

cc: W. T. Russell, Administrator, Region I, USNRC
T. P. Johnson, NRC Resident Inspector
E. M. Kelly, Senior Resident Site Inspector

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