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March 11, 1993
Docket No. 50-353
License No. NPF-85

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

SUBJECT: Licensee Event Report
Limerick Generating Station - Unit 2

This LER reports various automatic Engineered Safety Feature actuations resulting from a spurious Division 2 Loss of Coolant Accident (LOCA) signal. The inadvertent LOCA signal occurred after a technician inadvertently opened isolation valves during the performance of a surveillance test procedure.

Reference:	Docket No. 50-353
Report Number:	2-93-003
Revision Number:	00
Event Date:	February 13, 1993
Report Date:	March 11, 1993
Facility:	Limerick Generating Station P.O. Box 2300, Sanatoga, PA 19464-2300

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

Very truly yours,

J Doering

DMS:cah

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

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JE22

NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/88

LICENSEE EVENT REPORT (LER)

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TITLE (4) Engineered Safety Feature Actuations Resulting From a Spurious Division 2 Loss of Coolant Accident Signal due to 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 2	1 3	9 3	9 3	— 0 0 3 —	0 0	0 3	1 1	9 3		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1
POWER LEVEL (10): 1 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	
	AREA CODE	
G. J. Madsen, Regulatory Engineer, Limerick Generating Station	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)

		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)

On February 13, 1993, while the Unit was shutdown for refueling, an inadvertent Division 2 Loss of Coolant Accident (LOCA) signal was initiated. The LOCA signal was generated during the performance of a surveillance test, when an Instrumentation and Controls (I&C) technician was self-verifying instrument valve positions for a reactor water level differential pressure (dP) transmitter. The I&C technician inadvertently opened, then quickly re-closed, two isolation valves to the high pressure side reference leg head chamber. The two valves are utilized for calibration purposes only. The I&C technician mistakenly thought that the two valves were supposed to be open. The inadvertent LOCA signal caused various plant systems and automatic Engineered Safety Features to actuate automatically as designed. The actuations included the start of the 'B' Core Spray (CS) pump and subsequent injection, opening of the 'B' Low Pressure Coolant Injection Valve, the start of the D22 Emergency Diesel Generator, and various Primary Containment valve isolations. All systems and components functioned as designed in response to the inadvertent LOCA signal. All isolation signals were reset and the affected systems were expeditiously restored, thereby preventing any adverse impact on other plant systems. The cause of this event was cognitive personnel error. The corrective actions include counseling of the I&C technician, and dissemination of management expectations to all I&C personnel.

Unit Conditions Prior to the Event:

Unit 2 Operational Condition (OPCON) was 5 (Refueling) at 0% power level with the Reactor Pressure Vessel (RPV) pressure at 0 psig. The reactor cavity was flooded at 645 inches above the top of active fuel.

The 'B' Residual Heat Removal (RHR) system (EIIS:BO) was in service in the shutdown cooling mode of operation. There were no structures, systems or components out of service which contributed to this event.

Description

On February 13, 1993, at 0318 hours, while the unit was in shutdown for refueling, an inadvertent Division 2 Loss of Coolant Accident (LOCA) signal was initiated. The LOCA signal was generated during the performance of the Surveillance Test (ST) procedure ST-2-042-402-2, "Feedwater Main Turbine Trip System Actuation - Reactor Vessel Water Level - High; Level 8, Channel B Calibration/Functional Test." The calibration/functional sections of the ST procedure were successfully completed. An Instrumentation and Controls (I&C) technician was self-verifying instrument valve positions for an RPV water level differential pressure (dP) transmitter prior to allowing another I&C Technician to perform the Independent Verification of Restoration (IVOR) section of the ST procedure. While the dP transmitter was in service, the I&C technician inadvertently opened, then immediately re-closed, two isolation valves to the high pressure side reference leg head chamber (see Figure 1). The two isolation valves are used for calibration purposes only, and their normal positions are correctly indicated in the ST procedure. The I&C technician believed he had incorrectly left the two valves in the closed position following the performance of the ST procedure. Instead of rechecking the normal positions of the valves, the I&C technician mistakenly re-opened the two isolation valves. The resultant surge in reference leg pressure caused by quickly reclosing the isolation valves caused an inadvertent RPV low water level signal below Level 1 (+32 inches above the top of active fuel).

Main Control Room (MCR) Operations personnel immediately recognized the Division 2 LOCA signal. During the first few minutes of the event, the exact cause of the initiation was unknown, since the only two inputs that cause a LOCA signal are 1.68 psig drywell pressure or RPV water level below Level 1. MCR Operations personnel were unaware of the valving error made by the I&C technician. An actual primary containment pressure of 1.68 psig was ruled out because both equipment access hatches were removed, thereby exposing the primary containment to atmospheric pressure. An actual RPV water level below Level 1 was also ruled out because the water level was verified to be normal on the Shutdown Range water level indicator. Operations personnel implemented the Special Event (SE) procedure SE-10, "LOCA", in response to the inadvertent LOCA signal. Within minutes following this event, the I&C technician notified MCR Operations personnel of the valving error.

The inadvertent Division 2 LOCA signal caused the following Engineered Safety Features (ESF) to actuate as designed.

- o The 'B' Core Spray (CS) (EIIS:BM) pump started, and injected a total of approximately 2500 gallons of suppression pool water into the RPV for thirty seconds. Operations personnel secured the 'B' CS pump at 0319 hours.

- o The 'B' RHR pump was operating in the shutdown cooling mode of operation prior to the LOCA signal. After the LOCA signal initiated, the Low Pressure Coolant Injection (LPCI) injection valve opened, and the RHR Heat Exchanger bypass valve opened. The 'B' RHR pump continued to take suction from the RPV. At 0319 hours, Operations personnel closed the LPCI injection valve, throttled closed the RHR Heat Exchanger bypass valve, and returned the 'B' RHR pump to the shutdown cooling mode of operation.

- o The D22 Emergency Diesel Generator (EDG) (EIIS:EK) started, and continued to operate unloaded until 0358 hours, at which time Operations personnel secured the D22 EDG.

- o The Primary Containment and Reactor Vessel Isolation Control System (PCRVICS) was partially actuated resulting in the closure of the Containment Atmospheric Control sample valves SV-57-232, 234, 250, and 281.

- o The PCRVICS partial actuation also caused one of the two subsystems of the Containment Leak Detection System to isolate.

The General Plant (GP) procedure GP-8, "Primary and Secondary Containment Isolation Verification and Reset", was executed by Operations personnel to reset the Division 2 LOCA isolation signals. All isolation signals were reset, and the affected systems were restored by 0358 hours, utilizing appropriate plant procedures.

A four hour notification was made to the NRC at 0647 hours, on February 13, 1993, in accordance with the requirements of 10CFR50.72(b)(2)(ii), since this event resulted in various automatic actuations of ESFs. This report is being submitted in accordance with the requirements of 10CFR50.73(a)(2)(iv).

Analysis of the Event:

All systems and components functioned as designed in response to the spurious LOCA signal. The PCRVICS isolation signals were reset and the affected systems were expeditiously restored by Operations personnel, thereby preventing any adverse impact on other plant systems. There were no adverse consequences as a result of the PCRVICS isolations, since the Primary Containment was open to atmosphere and the affected systems were not required for operation. Procedure ST-2-042-204-2 is only performed during refueling operations, therefore this event could not have occurred under normal power operation. There was no release of radioactive material to the environment as a result of this event.

As a result of the 'B' CS pump injecting into the RPV for thirty seconds, inventory increased in the RPV causing the spent fuel pool skimmer surge tank located on elevation 352' of the refuel floor to overfill. Due to the overfill, RPV water leaked past the air and fire floor penetration seal for the skimmer surge tank and dripped down into Room 641A "Alcove," which is located on elevation 313' of the Reactor Enclosure. The RPV water then drained into appropriate floor drains inside the room. The normally locked closed Alcove room contains only concrete walls and piping, and plant personnel do not access this room. There are no valves or operating equipment located inside this room. Health Physics (HP) personnel identified that the room was contaminated, however, no plant personnel were contaminated as a result of the leakage.

During the one minute in which the 'B' RHR Shutdown Cooling system was not in service (i.e., while the LPCI injection valve and the RHR Heat Exchanger bypass valve were open), reactor coolant temperature did not increase and remained at 82 degrees Fahrenheit. The maximum reactor coolant temperature allowed by Technical Specifications (TS) is 140 degrees while in OPCON 5. Operations personnel had sufficient time to restore the 'B' RHR Shutdown Cooling system prior to exceeding the TS limit.

Immediate and follow-up actions for this type of event are provided in procedures SE-10 and GP-8. Licensed MCR operators receive requalification training to review and practice responses to simulated plant transients of this type. This training reinforces immediate operator actions, minimizing the time that the systems are isolated, and reducing the impact on the plant. Therefore, as a result of this adequate procedural guidance, training, and prompt operator actions, the consequences of this type of event are minimized.

Cause of the Event:

The cause of this event was cognitive personnel error. The normal positions of the two high pressure side reference leg head chamber isolation valves (see Figure 1) are correctly indicated in the ST procedure. The I&C technician had successfully completed the calibration/functional sections of the ST procedure and was performing a self-verification of his 'as left' valve positions. The I&C technician believed that he had incorrectly left the two isolation valves in the closed positions following the performance of the ST procedure. Instead of rechecking the normal positions of the isolation valves (i.e., closed), the I&C technician mistakenly re-opened, then quickly reclosed, the isolation valves causing the spurious Division 2 LOCA signal to initiate.

Corrective Actions

1. On February 13, 1993, a special shift briefing meeting was held for the oncoming shift of I&C technicians to discuss this event, the consequences of the event, and the reinforcement of Management's expectations concerning self-verification.

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

2. On February 16, 1993, a letter was issued to all I&C personnel describing the event, the consequences of the event, and a training overview on proper instrument valving techniques and the impact of valve mispositioning.

3. On February 23, 1993, the I&C technician involved in this event was counseled regarding the lack of adherence to self-checking/self-verification expectations.

4. The 'lessons learned' from this event will be included in the ongoing I&C Instrument Valving Course Number 40100.

5. An 'On the Job Training' module is expected to be implemented by April 1, 1993, to reinforce and enhance the expectations for the I&C technician's performance of self-checking/self-verification. Specific direction will be given on what actions should be taken when self-verification identifies an unexpected equipment status.

6. The locked door leading into Room 641A "Alcove" was sealed and posted as a 'High Radiation Area' by HP personnel on February 13, 1993, to prevent the 'contained' contamination from spreading. Decontamination of this room has been planned, and will be performed following completion of the Unit 2 refueling outage.

Previous Similar Occurrences:

LERs 1-84-007, 1-84-019, 1-85-037, 1-85-040, 1-85-046, 1-86-023, 1-88-025, and 1-91-005 reported I&C events caused by instrument valving errors. The majority of these events occurred just after Unit 1 initial criticality and resulted from an inadequate understanding or training of the effects of instrument valving techniques. Only one event has occurred since 1988 (i.e., LER 1-91-005). As a result of LER 1-84-019, I&C personnel developed an "Instrument Valving Course". All new I&C technicians receive this course. This course has been conducted every year since 1984. I&C technicians are then required to requalify this course on a two year frequency.

The cause of this event was cognitive personnel error due to the failure of the I&C technician to properly utilize self-verification techniques. We conclude that the previous development and implementation of the Instrument Valving Course has significantly reduced the frequency of occurrence of LERs caused by I&C valving errors. However, the additional corrective actions that have been taken as a direct result of this event will reinforce the expected performance during self-verification. Therefore, no additional actions are warranted.

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

FIGURE 1