

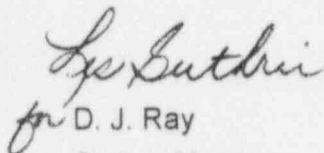
Commonwealth Edison Company
LaSalle Generating Station
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Marseilles, IL 61341-9757
Tel 815-357-6761



December 21, 1995
United States Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Licensee Event Report #95-011-00, Docket #050-374 is being submitted to your office in accordance with 10CFR50.73 (a) (2) (iv).

Sincerely,


for D. J. Ray
Station Manager
LaSalle County Station

Enclosure

cc: H. J. Miller, NRC Region III Administrator
K.D. Ihnen, Acting NRC Senior Resident Inspector
R. J. Zuffa, IDNS Resident Inspector
F. Niziolek, IDNS Senior Reactor Analyst
INPO - Records Center
D. L. Farrar, Nuclear Regulatory Services Manager

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LICENSEE EVENT REPORT (LER)

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNB8 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

FACILITY NAME (1) LaSalle County Station Unit Two	DOCKET NUMBER (2) 05000374	PAGE (3) 1 of 4
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TITLE (4)
Inadvertent ESF Actuation and Reactor Core Isolation Cooling Isolation 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 NAME	DOCKET NUMBER
11	27	95	95	011	00	12	21	95	NONE	
									FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)								
		20.2201(b)		20.2203(a)(3)(i)		50.73(a)(2)(iii)		73.71(b)		
POWER LEVEL (10)	100	20.2203(a)(1)		20.2203(a)(3)(ii)		X 50.73(a)(2)(iv)		73.71(c)		
		20.2203(a)(2)(i)		20.2203(a)(4)		50.73(a)(2)(v)		OTHER		
		20.2203(a)(2)(ii)		50.36(c)(1)		50.73(a)(2)(vii)		(Specify in Abstract below and in Text, NRC Form 366A)		
		20.2203(a)(2)(iii)		50.36(c)(2)		50.73(a)(2)(viii)(A)				
		20.2203(a)(2)(iv)		50.73(a)(2)(i)		50.73(a)(2)(viii)(B)				
		20.2203(a)(2)(v)		50.73(a)(2)(ii)		50.73(a)(2)(x)				

LICENSEE CONTACT FOR THIS LER (12)

NAME Russell Gremchuk, System Engineer	TELEPHONE NUMBER (Include Area Code) (815) 357-6761 x 2809
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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
A	RI	103AS-B203NX-JITTX6	S382	Yes					

SUPPLEMENTAL REPORT EXPECTED (14)

YES (If yes, complete EXPECTED SUBMISSION DATE)	X	NO	EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
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ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines 16)

At 0149 hours on November 27, 1995, during the performance of surveillance LIS-RI-401, "Unit 2 Steam Line High Flow RCIC Isolation Functional Test", the inboard steam line isolation valve (2E51-F063) closed on an invalid ESF signal. During this surveillance, steam flow differential pressure switch (PDS 2E31-N013BA) had failed a pressure test indicating a loss of integrity for an internal diaphragm. This switch actuates on a high flow condition in the RCIC steam line and provides an isolation signal to the inboard steam line isolation valve, 2E51-F063. The switch was declared inoperable and its transmitter isolated to ensure no spurious actuation would result. At 0149 hours, the Unit 2 Nuclear Station Operator (NSO) directed the operator stationed at the breaker panel to close the breaker for 2E51-F063. When the breaker was closed, the inboard steam line isolation valve (2E51-F063) closed due to the Unit 2 NSO's failure to verify that the RCIC high flow isolation logic signal had been reset. At 0154 hours, after determining the reason for the valve closing, the Unit 2 NSO reset the isolation logic and reopened 2E51-F063.

The cause of this event is attributable to a failure to follow procedure, inadequate procedural steps, and personnel errors by the Instrument Mechanic and the Unit 2 NSO.

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LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

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If more space is required, use additional copies of NRC Form 366A) (17)

PLANT AND SYSTEM IDENTIFICATION

General Electric - Boiling Water Reactor

Energy Industry Identification System (EIIS) codes are identified in the text as [XX].

A. CONDITION PRIOR TO EVENT

Unit(s): 2 Event Date: 11/27/95 Event Time: 0149 Hours

Reactor Mode(s): 1 Modes(s) Name: Run Power Level(s): 100%

B. DESCRIPTION OF EVENT

At 0149 hours on November 27, 1995, during the performance of surveillance LIS-RI-401, "Unit 2 Steam Line High Flow RCIC Isolation Functional Test", the inboard steam line isolation valve (2E51-F063) closed on an invalid ESF signal. During this surveillance, steam flow differential pressure switch (PDS 2E31-N013BA) had failed a pressure test indicating a loss of integrity for an internal diaphragm. The switch was declared inoperable and its transmitter isolated to ensure no spurious actuation would occur. This switch actuates on a high flow condition in the Reactor Core Isolation Cooling (RCIC, RI)(BN) steam line and provides an isolation signal to the inboard steam line isolation valve (2E51-F063). This motor operated valve is normally open and is prevented from closing during this surveillance by opening the circuit breaker in the electrical power supply to the RCIC inboard steam valve. At 0149 hours the Unit 2 NSO directed the Operator stationed at the breaker panel to close the circuit breaker for 2E51-F063. After the breaker was closed, valve 2E51-F063 went closed unexpectedly due to the failure to reset the RCIC high flow isolation logic signal at the main control board.

At 0154 hours, after determining the reason for the valve closing, the Unit 2 NSO reset the isolation logic and reopened 2E51-F063.

C. CAUSE OF EVENT

When steam flow differential pressure switch (PDS 2E31-N013BA) failed the diaphragm integrity test, the Instrument Mechanic could not complete LIS-RI-401 as written. The procedure requires returning the differential pressure transmitter to normal alignment

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before completing the remaining steps which reset isolation logic and re-close the inboard steam line isolation valve (2E51-F063) breaker. No clear procedural guidance existed to give the Instrument Mechanic and NSO an Emergency Exit process (success Path). After noting the failure and appropriately stopping the remaining actions in the procedure, the Instrument Mechanic made a personnel error in not instructing the NSO to close the circuit breaker.

The Instrument Mechanic used only one of the available indicators in determining if the isolation logic had been reset. Distinguishing between alarms in slow or normal flash without also using the reset keylock switch position or indicating light to verify reset was inadequate.

The Unit 2 NSO did not verify that isolation logic had been reset prior to having the breaker closed. LIS-RI-401 is performed by having the Instrument Mechanic follow the procedure and instruct the NSO when to reset isolation logic and close breakers. Three transmitters had been previously tested satisfactorily. The Assist NSO, not the Unit 2 NSO, had performed the previous resets when instructed by the Instrument Mechanic.

The Instrument Mechanic and the Unit 2 NSO both had become distracted when the steam flow differential pressure switch (PDS 2E31-N013BA transmitter failed. At this time, their actions were directed to assisting the shift supervisors in determining the effect on RCIC system operability, getting the transmitter realigned/isolated, documenting the failure as requested by the Shift Engineer, and instructing the Instrument Mechanics and Operators in the plant.

The Instrument Mechanic did not review the remaining, open steps of LIS-RI-401 to determine the course of action required to complete the procedure after the transmitter failure. The procedure includes guidance to notify supervision.

This is reportable per 10CFR50.73(a)(2)(iv) due to an automatic actuation of an engineered safety feature (ESF).

D. ASSESSMENT OF SAFETY CONSEQUENCES

The safety significance of this event was minimal. Primary containment isolation was maintained because the outboard (Division 1) steam line isolation valve was available and had just been successfully tested. High Pressure Core Spray (HPCS) was available as

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a high pressure reactor vessel injection system, the Automated Depressurization System (ADS) and all low pressure Emergency Core Cooling System (ECCS) systems were available throughout the event. The steam flow differential pressure switch (PDS 2E31-NO13BA passed the set point calibration surveillance satisfactorily, but with a torn diaphragm had the potential to not operate as designed to provide steam line isolation in the event of a line break.

E. CORRECTIVE ACTIONS

The failed differential pressure switch was immediately replaced.

The Instrument Mechanic and the Unit 2 NSO were counseled on procedural adherence and the need to perform an adequate self-check during the performance of surveillances.

This event will be used to develop case study training on the lessons-learned for presentation to Nuclear Station Operators and Instrument Mechanics regarding: Proper self-check and three-way communications and RCIC isolation systems and logic.

The Instrument Maintenance Department will conduct an evaluation of their surveillance procedures and incorporate an Emergency Exit process where applicable. These changes will be implemented during the procedure revisions for Improved Technical Specifications. In the interim, LIP-GM-902, "General Requirements for Performance of Instrument Maintenance Department Procedures", will be revised to include an Emergency Exit process to prevent recurrence of this event.

F. PREVIOUS OCCURRENCES

LER NUMBER	TITLE
374/93-001-00	RCIC High Flow Isolation Static-O-Ring (SOR) Failure Due to a Torn Diaphragm
373/90-011-00	Failure of RCIC Steam Line High Flow SOR Differential Pressure Switch
373/90-009-00	Failed RCIC Hi Steam Flow DP Switch Due to Torn Diaphragm/Unknown

G. COMPONENT FAILURE DATA

Static-O-Ring differential pressure switch, model 103AS-B203NX-JJTTX6