LICENSEE EVENT REPORT (LER) Form Rev 2.0

Facility Name (1)	Docket Number (2)	Page (3)
LaSalle County Station Unit 1	0 5 0 0 0 3 7 3	1 of 0 4

Title (4)

Reactor Core Isolation Cooling Hi Steam Flow Isolation Switch Failed Diaphragm

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 0 0 0
0 3	0 9	8 9	8 9	0 1 2	0 0	0 4	0 7	8 9		0 5 0 0 0

OPERATING MODE (9)	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR (Check one or more of the following) (11)					
1	___ 20.402(b)	___ 20.405(c)	___ 50.73(a)(2)(iv)	___ 73.71(b)		
POWER LEVEL (10) 1 0 0	___ 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)	X Other (Specify in Abstract below and in Text) Voluntary		
	___ 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
Michael Tennyson, Technical Staff Engineer, extension 2704	AREA CODE 8 1 5 3 5 7 - 6 7 6 1

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
X	R I	F S	S 3 8 2	Y					

SUPPLEMENTAL REPORT EXPECTED (14)		Expected Submission Date (15)	Month	Day	Year
X Yes (If yes, complete EXPECTED SUBMISSION DATE)	___ NO		0 6	3 0	8 9

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

On March 9, 1989, at 2045 hours, during performance of LaSalle instrument surveillance LIS-RI-101, "Unit 1 Steam Line High Flow Reactor Core Isolation Cooling (RCIC) Isolation Calibrational Test," Pressure Differential Switch PDS-1E31-N013BB was found to have a diaphragm leak. Unit 1 was in Operational Condition 1 (Run) at 100% power level. The setpoint for this switch was found within the action limit and the Limiting Condition for Operation (LCO).

This switch functions with similar switch PDS-1E31-N013BA, to provide Division II (Inboard) isolation of the RCIC/Residual Heat Removal Steam Line and to initiate a RCIC turbine trip.

1E31-N013BB is connected in reverse parallel to 1E31-N013BA. The design function of 1E31-N013BB is to initiate an inboard containment isolation in the event of an instrument line break. Particularly a break in the instrument line coming from the high pressure side of the steam flow elbow leading to the high pressure side of 1E31-N013BA and to the low pressure side of 1E31-N013BB. RCIC system had been declared inoperable on March 9, 1989 at 1300 hours in order to perform the required surveillance. The High Pressure Core Spray system remained operable throughout the duration of this event.

A replacement switch was installed, calibrated and functionally tested satisfactorily. RCIC system was declared operable on March 10, 1989 at 1940 hours. The replaced switch will be disassembled and inspected in an attempt to determine root cause.

This event is reported to the Nuclear Regulatory Commission as a Licensee Event Report in accordance with the requirements of I.E. Bulletin 86-02, "Static-O-Ring (SOR) Differential Pressure Switches."

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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): 1 Event Date: 3/9/89 Event Time: 2045 Hours

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

B. DESCRIPTION OF EVENT

Reactor Core Isolation Cooling (RCIC, RI) [BN] Steam Line High Flow Isolation (PC) [JM] Switch PDS-1E31-N013BB was found to have a diaphragm leak on March 9, 1989 at 2045 hours. The problem was noted while a calibrational test was being performed per LaSalle Instrument Surveillance LIS-RI-101, "Unit 1 Steam Line High Flow RCIC Isolation Calibration Test." This surveillance calls for verifying that the diaphragm does not leak in addition to verifying that the proper actuations occur at the proper setpoints. During this surveillance, the Instrument Maintenance Technician (CST) found the setpoint within the Action Limit and Limiting Condition for Operation (LCO) for this instrument. RCIC had been declared inoperable per Degraded Equipment Log (DEL) Number 90-89-1, prior to the start of the surveillance.

PDS-1E31-N013BB works in parallel with differential switch PDS-1E31-N013BA to trip the RCIC turbine and automatically close the "RCIC Steam Line Inboard Isolation Valve" (1E51-F063) and the "RCIC Steam Line Warmup Valve" (1E51-F076). A high differential pressure condition for either PDS-1E31-N013BA or PDS-1E31-N013BB will cause an actuation to take place.

1E31-N013BB is connected in reverse parallel to 1E31-N013BA. The design function of 1E31-N013BB is to initiate an inboard containment isolation in the event of an instrument line break. Particularly a break in the instrument line coming from the high pressure side of the steam flow elbow leading to the high pressure side of 1E31-N013BA and to the low pressure side of 1E31-N013BB.

No other inoperable equipment/systems contributed to this event. No automatic or manual safety system actuations occurred and none were required. No Operator actions contributed to the causation or severity of this event. Actions taken to correct the cause of this event were timely and appropriate.

This event is reported to the Nuclear Regulatory Commission as a Licensee Event Report in accordance with the requirements of I.E. Bulletin 86-02, "Static-O-Ring (SOR) Differential Pressure Switches."

C. APPARENT CAUSE OF EVENT

The root cause of this diaphragm failure can not be readily determined at this time. The switch has been sent to SOR Inc. for inspection and to determine cause of failure.

The diaphragm divides the two halves of the differential pressure switch into a "high" side and a "low" side. As system flow increases, the difference in pressures between the sides of the switch also increases. This causes the diaphragm to flex. A piston assembly is affixed to the diaphragm and drives a microswitch which trips when the deflection of the diaphragm is sufficient.

D. SAFETY ANALYSIS OF EVENT

A leaking diaphragm of PDS-1E31-N013BB would allow for the equalization of pressure between the high and low instrument lines. This could prevent switch PDS-1E31-N013BA (which uses the same instrument lines as PDS-1E31-N013BB) from sensing any high steam flow condition, or alter its trip setpoint, rendering the switch inoperable. However, when calibrated, the switch did trip at an acceptable trip setpoint. In addition redundant equipment (pressure differential switches PDS-1E31-N013AA and PDS-1E31-N013AB) remained fully operable, and would have provided the outboard isolation of valve 1E51-F008, RCIC Outboard Isolation Valve.

In addition, both the inboard and outboard isolation functions for Residual Heat Removal (RHR, RH) [BO] steam line high flow would have occurred as designed had a high flow condition existed in the RHR steam line downstream of the RHR Heat Exchanger Outboard Isolation Valve, 1E51-F064.

The corrective action for this event was performed as required by Technical Specifications. This resulted in RCIC being declared inoperable. During this time, however, High Pressure Core Spray (HPCS, HP) [BG] was operable.

E. CORRECTIVE ACTIONS

Inboard isolation valves 1E51-F063 and 1E51-F076 were secured closed prior to the start of the surveillance. The RCIC high flow isolation was declared inoperable and the isolation valves remained closed.

RCIC was entered as inoperable in the Degraded Equipment Log (DEL) Number 90-89-1. Action Statement b. of Technical Specification 3.7.3 allows fourteen days to restore operability to RCIC, provided HPCS remains operable.

E. CORRECTIVE ACTIONS (Continued)

A new SOR differential pressure switch identical to the one that failed was certified for use in the RCIC steam line high flow application using LIP-GM-952, "Static-O-Ring Differential Pressure Switch Operability Test," and LIP-GM-956, "Analysis of Static-O-Ring Differential pressure Switch Data." This new switch was installed per LIP-GM-946, "Installation Procedure for S-O-R Model 103/102 Environmentally Qualified Differential Pressure Switches," under LaSalle Work Request L88180 on March 10, 1989. The switch was calibrated per LIS-RI-101, "Unit 1 Steam Line High Flow RCIC Isolation Calibration." Operability was restored to RCIC on March 10, 1989 at 1940 hours.

The failed Differential Pressure Switch, PDS-1E31-N013BB, will be sent to SOR Inc. where it will be disassembled and inspected. The findings of this inspection will be included in a supplement to this Licensee Event Report and tracked by Action Item Record (AIR) 373-200-89-03001.

F. PREVIOUS EVENTS

LER Number	Title
374/86-018-01	Failure of Reactor Core Isolation Cooling Steam Line Flow Isolation Switch Due to Torn Diaphragm
374/87-016-01	Defective Low Pressure Core Spray Minimum Flow Switch
374/87-019-01	Failure of Static-O-Ring Differential Pressure Switch Due to Leakage Across Diaphragm
373/88-009-01	High Pressure Core Spray Low Low Level Initiation Static-O-Ring Level Switch Diaphragm Rupture

G. COMPONENT FAILURE DATA

Manufacturer	Nomenclature	Model Number	MFG Part Number
SOR, Inc.	Differential Pressure Switch	103AS-B203-NX JJTTX6	N/A

Commonwealth Edison
LaSalle County Nuclear Station
Rural Route #1, Box 220
Marseilles, Illinois 61341
Telephone 815/357-6761

April 7, 1989

Director of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Mail Station P1-137
Washington, D.C. 20555

Dear Sir:

Licensee Event Report #89-012-00, Docket #050-373 is being submitted as a Voluntary Report to your office in accordance with NRC I.E. Bulletin 86-02, "Static-O-Ring Differential Pressure Switches."

G. J. Diederich for
Station Manager
LaSalle County Station

GJD/MMT/kg

Enclosure

xc: Nuclear Licensing Administrator
NRC Resident Inspector
NRC Region III Administrator
INPO - Records Center

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