LICENSEE EVENT REPORT (LER)

Form Rev 2.0

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

Title (4)

Failure of Reactor Core Isolation Cooling Steam Line High Flow Isolation Switch Due to 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 8	1 7	8 8	8 8	0 0 9	0 0	0 9	1 6	8 8		0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 0 7 7

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR (Check one or more of the following) (11)

20.402(b)	20.405(c)	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)	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	
	AREA CODE	
Alan J. McLaughlin, Technical Staff Engineer, extension 576	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	B N	D P 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		1 2	3 1	8 8

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

On August 17, 1988, at 1030 hours, during performance of LaSalle Instrument Surveillance LIS-RI-401, "Unit 2 Steam Line High Flow RCIC Isolation Functional Test," Pressure Differential Switch PDS-2E31-N007BB was found to have an apparent ruptured diaphragm. Unit 2 was in Operational Condition 1 (Run) at 77% power.

This switch functions with similar switch, PDS-2E31-N007BA, to provide Division II (Inboard) isolation of the Reactor Core Isolation Cooling (RCIC)/Residual Heat Removal (RHR) Steam line and to initiate a RCIC turbine trip under a high flow (line break) condition. However, separate switches PDS-2E31-N007AA and PDS-2E31-N007AB were still available to provide Division I (outboard) isolation and turbine trip.

RCIC system had been declared inoperable on August 17, 1988 at 0730 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. The replaced switch will be disassembled and inspected in an attempt to determine root cause. The new switch and the RCIC system were then believed to be operable and declared, as such, on August 18, 1988 at 1100 hours. However, on September 1, 1988 at 1631 hours this switch actuated due to not having been fully valved in on August 18, 1988.

The September 1, 1988 spurious actuation will be reported per Licensee Event Report 374/88-011-00.

This event is reported to the NRC as a Voluntary Report and the requirements of IE Bulletin 86-02, "Static-O-Ring Differential Pressure Switches."

IE22

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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): 2 Event Date: 8/17/88 Event Time: 1030 Hours

Reactor Mode(s): 1 Mode(s) Name: Run Power Level(s): 77%

B. DESCRIPTION OF EVENT

Reactor Core Isolation Cooling (RCIC, RI), [BN] Steam Line High Flow Isolation (PC) [JM] Switch PDS-2E31-N007BB was found to be nonfunctional on August 17, 1988 at 1030 hours. The problem was noted while a functional test was being performed per LaSalle Instrument Surveillance LIS-RI-401, "Unit 2 Steam Line High Flow RCIC Isolation Functional Test," at Step F.65. This step calls for applying a differential pressure to switch PDS-2E31-N007BB to check that it trips properly and provides the necessary actuations. During this step, the Instrument Maintenance Technician (CST) could not get this instrument to trip. Further investigation revealed that the diaphragm was leaking.

PDS-2E31-N007BB works in parallel with differential switch PDS-2E31-N007BA to trip the RCIC turbine and automatically close the "RCIC Steam Line Inboard Isolation Valve" (2E51-F063) and the "RCIC Steam Line Warmup Valve" (2E51-F076). A condition for either PDS-2E31-N007BA or PDS-2E31-N007BB will cause an actuation to take place.

The design function of these switches is to sense high steam flow during operation of the Residual Heat Removal (RHR, RH) [BO] system in Steam Condensing Mode and initiate an inboard containment isolation.

No other inoperable equipment/systems contributed to this event. No automatic or manual safety 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.

However, due to the new switch not being fully valved in, a spurious actuation occurred on September 1, 1988 at 1631 hours, the details of which will be reported per Licensee Event Report 374/88-011-00.

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

C. APPARENT CAUSE OF EVENT

PDS-2E31-N007BB apparently failed to function because of a torn diaphragm inside the differential pressure switch.

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.

The root cause of this diaphragm failure can not be readily determined at this time.

D. SAFETY ANALYSIS OF EVENT

During the time that PDS-2E31-N007BB was inoperable, the inboard isolation function for the Steam Condensing Mode of RHR could not have occurred as designed. The torn diaphragm of PDS-2E31-N007BB would allow for the equalization of pressure between the high and low instrument lines. This could prevent switch PDS-2E31-N007BA (which uses the same instrument lines as PDS-2E31-N007BB) from sensing any high steam flow condition, or alter its trip setpoint, rendering the switch inoperable also. However, redundant equipment (pressure differential switches PDS-2E31-N007AA and PDS-2E31-N007AB) remained fully operable, and would have provided the outboard isolation of valves 2E51-F064, "RHR Heat Exchanger Outboard Isolation Valve," and 2E51-F091, "RHR Heat Exchanger Warmup Valve."

In addition, both the inboard and outboard isolation functions for RCIC steam line high flow (to RCIC Turbine) would have occurred as designed had a high flow condition existed in the RCIC steam line downstream of the "RCIC Steam Line Outboard Isolation Valve," 2E51-F008.

Furthermore, the RCIC area temperature and differential temperature monitors were available throughout this event to sense any actual steam leak which may have occurred.

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

E. CORRECTIVE ACTIONS

Action Statement 22 of Technical Specification 3.3.2 was entered. This statement allows one hour to repair an inoperable isolation channel or the associated isolation valves must be secured in the isolated condition. Accordingly, valves 2E51-F063 and 2E51-F076 were secured closed per out-of-service 2-424-88.

This response rendered RCIC inoperable because there was no pathway for reactor steam (MS) [SB] to reach the RCIC turbine. RCIC was entered as inoperable in Degraded Equipment Log (DEL) Number 401-88-2. 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 failed one 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 10/102 Environmentally Qualified Differential Pressure Switches" under LaSalle Work Request L83041 on August 18, 1988. The switch was calibrated per LIS-RI-201, "Unit 2 Steam Line High Flow RCIC Isolation Calibration." The out-of-service on the inboard isolation valves was cleared and full operability was believed to be restored to RCIC on August 18, 1988 at 1100 hours.

However, due to the fact that the intermediate stop valves were left closed, the new switch was not properly valved in. This lead to the September 1, 1988 actuation which will be reported per Licensee Event Report 374/88-011-00.

The failed Differential Pressure Switch, PDS-2E31-N007BB, 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) 374-200-88-03501.

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
374/87-020-01	Failure of Several SOR Differential Pressure Switches Due to Diaphragm Ruptures
373/88-009-00	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	103-AS-B203-NX-C1A-JJ-TTX6	N/A

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

September 16, 1988

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

Dear Sir:

Licensee Event Report #88-009-00, Docket #050-374 is being submitted as a Voluntary Report to your office in accordance with I.E. Bulletin 86-02, "Static-O-Ring Differential Pressure Switches."

W R Huntington
for G. J. Diederich
Station Manager
LaSalle County Station

GJD/AJM/kg

Enclosure

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

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