LICENSEE EVENT REPORT (LER)

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

Title (4) Defective Low Pressure Core Spray Minimum Flow Switch

Event Date (5) Month	Day	Year	LER Number (6) Year	Sequential Number	Revision Number	Report Date (7) Month	Day	Year	Other Facilities Involved (8) Facility Names	Docket Number(s)
										0 5 0 0 0
0 7	1 3	8 7	8 7	0 1 6	0 1	0 1	2 7	8 8		0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 0 9 5

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	
Richard J. Rohrer, Technical Staff Engineer, ext. 250	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	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS
X	B M	P D S	S 3 8 2	Y					

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)

Low Pressure Core Spray (LPCS) minimum flow switch, FS-2E21-N004, was found to be leaking water through the switch internals during the performance of Instrument Surveillance LIS-LP-202, "Unit 2 LPCS Minimum Flow Bypass Calibration," on July 13, 1987, at 1045 hours. At the time of this event, Unit 2 was in Operational Condition 1 (Run) at 95% rated power. The leakage was very small and did not affect the operation of the switch as demonstrated during the calibration.

A Work Request was written to replace the defective switch with another one, like-for-like. The replacement switch was installed, calibrated and placed in service by 2300 hours on July 15, 1987. The leaking instrument has been disassembled and inspected by SOR, Inc. and Commonwealth Edison's System Materials Analysis Division. These inspections revealed two small slits in the switch's diaphragm. The cause of these defects could not be determined.

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

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LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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TEXT

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: 7/13/87 Event Time: 1045 hours

Reactor Mode(s): 1 Mode(s) Name: Run Power Level(s): 95%

B. DESCRIPTION OF EVENT

The Low Pressure Core Spray (LPCS) [BM] minimum flow switch (FS-2E21-N004) was found to be leaking water through the switch internals. This discovery was made during the routine performance of Instrument Surveillance LIS-LP-202, "Unit 2 LPCS Minimum Flow Bypass Calibration," on July 13, 1987, at 1045 hours. At the time, Unit 2 was in Operational Condition 1 (Run) at 95% rated power.

This switch functions to open the LPCS minimum flow bypass valve, 2E21-F011, on a low flow signal when the LPCS pump is running.

The calibration procedure calls for checking the leakage integrity of the switch by maintaining a differential pressure approximately equal to the switch setpoint (23 inches of water column) between the high and low pressure sides of the switch. The test configuration for this leakage integrity check is shown in Figure 1. When this leakage check was performed, the water level in the high side bottle was observed to slowly decrease while the water level in the low side bottle was observed to slowly increase. It was thought that this may have been due to a leaking equalizing valve, so the technician connected the water bottles directly to the instrument ports and checked for leakage again. The same behavior was observed, indicating leakage through the switch's internal components. Further testing was done to qualify the nature of the leak. The water bottles were left installed with approximately 1 inch of water column ("WC) differential pressure applied. After one hour, no change in water levels was observed. The differential pressure was then raised to approximately 27 "WC and a slow leakage was observed. No leakage was observed for an equal differential pressure applied in the reverse direction (i.e., from the low side to the high side of the switch).

From these findings it is clear that the defect had minimal impact on switch operation. This is supported by the fact that the switch passed its calibration test with an "As Found" setpoint exactly at the desired setpoint of 23.0 "WC. All switch behavior was as expected during the calibration and all actuations occurred as required.

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

TEXT

C. APPARENT CAUSE OF EVENT

This event was caused by a minor defect in the diaphragm of the switch.

The switch was sent to its manufacturer, SOR, Inc., for disassembly and inspection. In addition, Commonwealth Edison's System Materials Analysis Division (SMAD) has examined the switch. These inspections revealed two small slits in the diaphragm near a machined part of the switch's body. The mechanism which produced these tears could not be determined.

D. SAFETY ANALYSIS OF EVENT

This event had minimal impact on plant safety.

The instrument calibration, performed in conjunction with the switch integrity test, demonstrated that the switch was functioning properly. This instrument would have performed its design function.

The internal leakage discovered during this surveillance could possibly have increased to a point where switch actuation would not occur per design. This is unlikely, though, because this differential pressure switch, along with all others made by the same manufacturer, is being checked monthly for leakage integrity. Any defect permitting water to cross the switch diaphragm would be discovered at an early stage in its development, as it was in this case.

E. CORRECTIVE ACTIONS

Although the minimum flow switch was still capable of performing as designed, it was declared inoperable because of the internal defect. This made the LPCS system technically inoperable. Technical Specification 3.5.1 governs the operability requirements of Emergency Core Cooling Systems when the reactor is at power. Action Statement a.1 of this specification allowed seven days to restore operability to the LPCS System, since all other Emergency Core Cooling Systems were operable.

Work Request L70377 was written to replace the defective switch with another one, like-for-like. A new switch was certified for use in accordance with procedures LIP-GM-952, "Static-O-Ring Differential Pressure Switch Operability Test," and LIP-GM-956, "Analysis of Static-O-Ring Differential Pressure Switch Test Data." The replacement switch was installed, calibrated, and placed in service by 2300 hours on July 15, 1987. LPCS was then declared operable, well within the seven days allowed by Technical Specification 3.5.1.

Commonwealth Edison's BWR Engineering Department (BWRED) has evaluated the SOR and SMAD inspection results. BWRED has concluded that the failure of this diaphragm is random in nature and that the rate of SOR diaphragm failures at LaSalle is as low as or lower than the failure rate of any differential pressure sensing device used in the nuclear power industry.

The corrective actions taken to date to improve SOR's manufacturing processes and to perform monthly leakage integrity tests are sufficient, given the low failure rate of these devices. No further action is required.

TEXT

F. PREVIOUS EVENTS

LER Number	Title
374/86-018-01	Failure of RCIC Steam Line Isolation Switch Due to Torn Diaphragm

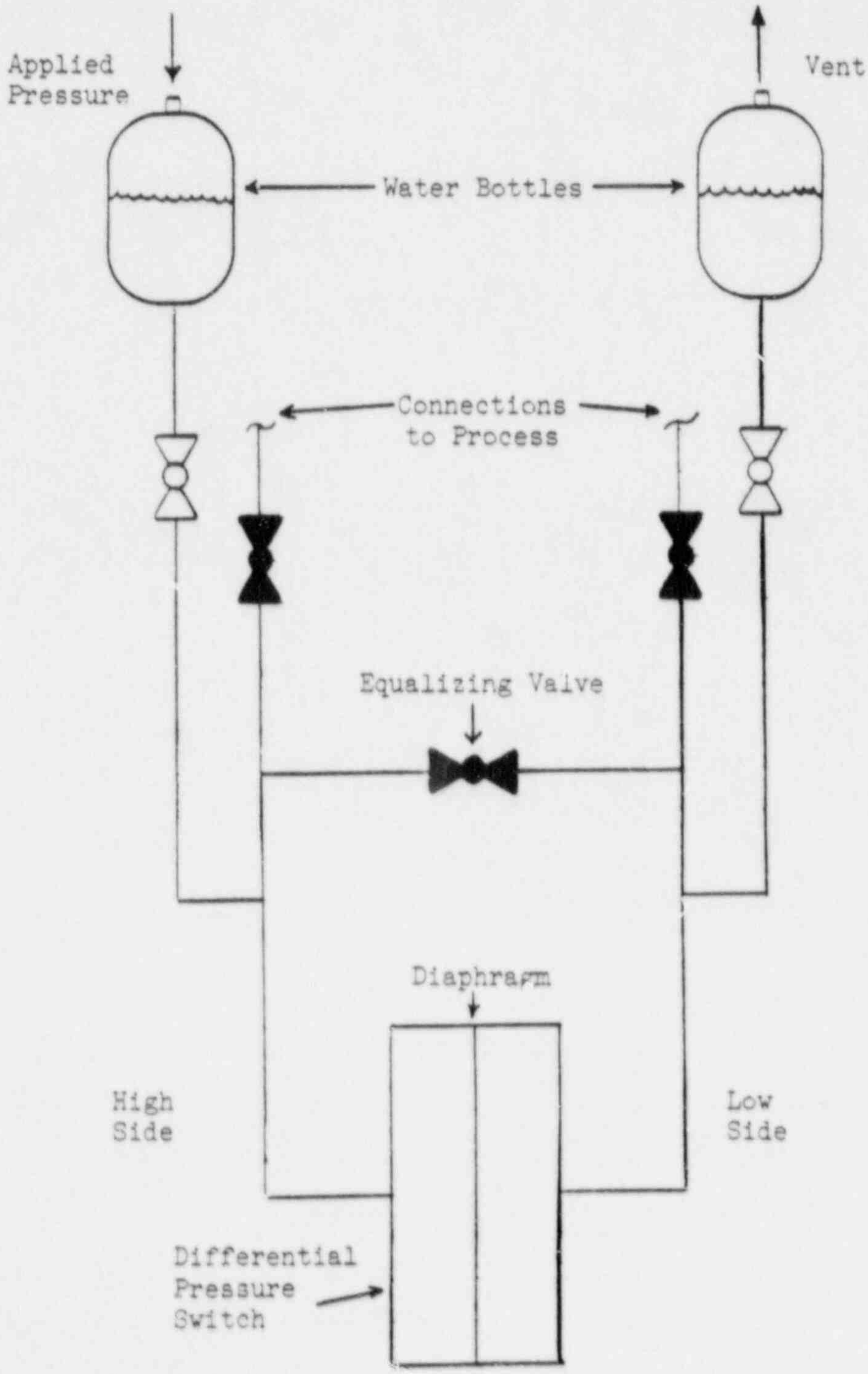
G. COMPONENT FAILURE DATA

Manufacturer	Nomenclature	Model Number	MFG Part Number
SOR, Inc.	Differential Pressure Switch	103-AS-B202-NX-C1A-JJ-TTX6	N/A

TEXT

FIGURE 1

CONFIGURATION OF TEST EQUIPMENT FOR LEAKAGE INTEGRITY CHECK



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

January 27, 1988

U.S. Nuclear Regulatory Commission
Document Contrc. Desk
Washington, D.C. 20555

Dear Sir:

Licensee Event Report #87-016-01, Docket #050-374 is being submitted to your office to supercede previously submitted Licensee Event Report 87-016-00 to revise report with results from manufacturer inspections.

for G. J. Diederich
Station Manager
LaSalle County Station

GJD/RJR/kg

Enclosure

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

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