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ComEd

October 2, 2000

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

LaSalle County Station, Unit 2
Facility Operating License No. NPF-18
NRC Docket No. 50-374

Subject: Licensee Event Report

In accordance with 10 CFR 50.73(a)(2)(iv), Commonwealth Edison (ComEd) Company is submitting Licensee Event Report Number 00-004-00, Docket No. 050-374.

Should you have any questions concerning this letter, please contact Mr. William J. Riffer, Regulatory Assurance Manager, at (815) 357-6761, extension 2383.

Respectfully,

Charles G. Pardee
Site Vice President
LaSalle County Station

Attachment: Licensee Event Report

cc: Regional Administrator - NRC Region III
NRC Senior Resident Inspector - LaSalle County Station

IE22

NRC FORM 366 (6-1998) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-
EXPIRES 06/30/2001

LICENSEE EVENT REPORT (LER)

Estimated burden per response to comply with this mandatory information collection request: 50 hrs. Reported lessons learned are incorporated into the licensing process and fed back to industry. Forward comments regarding burden estimate to the Records Management Branch (T-6 F33), 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. If an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

FACILITY NAME (1):	DOCKET NUMBER (2)	PAGE (3)
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TITLE (4) Reactor Core Isolation Cooling (RCIC) Isolation Due to RCIC High Steam Flow Indication

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
09	01	00	00	004	00	10	02	00	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	1
POWER LEVEL (10)	100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

☐ 20.2201(b)	☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)	☐ 50.73(a)(2)(viii)
☐ 20.2203(a)(1)	☐ 20.2003(a)(3)(i)	☐ 50.73(a)(2)(ii)	☐ 50.73(a)(2)(x)
☐ 20.2203(a)(2)(i)	☐ 20.2003(a)(3)(ii)	☐ 50.73(a)(2)(iii)	☐ 73.71
☐ 20.2203(a)(2)(ii)	☐ 20.2003(a)(4)	☒ 50.73(a)(2)(iv)	☐ OTHER
☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(1)	☐ 50.73(a)(2)(v)	Specify n Abstract below or in NRC Form 366A
☐ 20.2203(a)(2)(iv)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(vii)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Chuck Maney, Operations	(815) 357-6761 Extension 2929

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
☐ YES (If yes, complete EXPECTED SUBMISSION DATE)	☒ NO				

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

At 1420 hours on September 1, 2000, LaSalle Unit 2 was preparing to return Reactor Core Isolation Cooling (RCIC)(BN) to service in accordance with procedure LOP-RI-05, "Preparation for Standby Operation of the Reactor Core Isolation Cooling System." Fifteen seconds after opening the RCIC Warm-up Bypass valve 2E51-F076, a RCIC Steam Line Differential Pressure alarm was received and valve 2E51-F076 closed automatically due to a RCIC High Steam Flow containment isolation signal.

The root cause of the event was the presence of water on the upstream side of valve 2E51-F076. The water flashed into steam when valve 2E51-F076 was opened, resulting in a momentary high steam flow signal, which resulted in a containment isolation signal.

Procedure LOP-RI-05 will be revised to isolate the steam flow instrumentation from service until valves 2E51-F076 and RCIC Steam Line Outboard Isolation valve 2E51-F008 have both been opened. The safety significance of this event was minimal. The Emergency Core Cooling Systems were not challenged and were fully operational.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
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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, 3489 Megawatts Thermal Rated Core Power

A. CONDITION PRIOR TO EVENT

Unit(s): 2 Event Date: 09/01/00 Event Time: 1420 Hours
Reactor Mode(s): 1 Power Level(s): 100
Mode(s) Name: Run

B. DESCRIPTION OF EVENT

At 1420 hours on September 1, 2000, following maintenance on RCIC Steam Line Isolation valve 2E51-F008, LaSalle Unit 2 was preparing to return Reactor Core Isolation Cooling (RCIC)(BN) to service in accordance with procedure LOP-RI-05, "Preparation for Standby Operation of the Reactor Core Isolation Cooling System." Fifteen seconds after opening RCIC Warm-up Bypass valve 2E51-F076, a RCIC Steam Line Differential Pressure alarm was received and valve 2E51-F076 closed automatically due to a RCIC high steam flow containment isolation signal. All equipment responded to the isolation signal as expected. There were no inappropriate personnel actions.

During maintenance activities, water had accumulated on the upstream side of the closed 2E51-F076 valve. When valve 2E51-F076 was opened, the water flashed into steam and created a pressure wave that appeared as a high steam flow to the instrumentation, resulting in a containment isolation signal. The pressurizing and warming line is a one-inch line, and entrapped water would flash upon passing into the ten-inch down stream line. The duration of the pressure transient exceeded the time delay setting of 3.65 seconds for the 2H13-P618 Steam Line D/P (K48) relay, resulting in the Division 2 system isolation. The duration of the pressure transient did not exceed the time delay setting of 4.14 seconds for the Division 1 relay. Both division relays have a calibration range of three to five seconds.

This event is reportable in accordance with 10 CFR 50.73(a)(2)(iv), as an event that resulted in a manual or automatic actuation of any engineered safety feature (ESF), including the reactor protection system (RPS).

C. CAUSE OF EVENT

The root cause of the event was the presence of water on the upstream side of valve 2E51-F076 prior to opening the valve. The water comes from condensation when the RCIC steam line is isolated. The water flashed into steam and created a pressure wave that appeared as a high steam flow to the down stream flow instruments. The pressurizing and warming line is a one-inch line, and entrapped water would flash upon passing into a ten-inch down stream line. The duration of the pressure transient exceeded the time delay setting of the K48 relay, resulting in the spurious system isolation.

Water accumulation above the RCIC inboard isolation valves is an expected and proven condition. The system has been successfully repressurized numerous times utilizing the existing procedure revision and the as-found instrumentation settings. What is distinctive this time was the cooling of the 10" steam pipe by cycling valve 2E51-F008 while the system was isolated. Valve 2E51-F008 was cycled to consolidate the valve packing after maintenance.

D. SAFETY ANALYSIS

The safety significance of this event was minimal. The Emergency Core Cooling Systems were not challenged and remained fully operable. The primary containment remained fully operable. Control Room personnel took appropriate actions in response to valve 2E51-F076 actuation.

E. CORRECTIVE ACTIONS

Corrective Actions to Prevent Recurrence

The corrective action is to revise the procedure LOP-RI-05 to isolate the steam flow instrumentation from service until both valves 2E51-F076 and 2E51-F008 have been opened. This procedure is generic to both Unit 1 and Unit 2. Presently, the procedure returns the flow instrumentation to service after valve 2E51-F008 is opened. (ATM# 34563)

F. PREVIOUS OCCURRENCES

A review of LaSalle Licensee Event Reports over the previous five years found no previous occurrences of a RCIC isolation.

G. COMPONENT FAILURE DATA

Since no component failure occurred, this section is not applicable.