Commonwealth Edison
Dresden Nuclear Power Station
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March 7, 1994

GFS LTR: 94-0076

U.S. Nuclear Regulatory Commission
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License Event Report 94-003, Docket 50-237 is being submitted as required by Technical Specification 6.6, NUREG 1022 and 10 CFR 50.73 (a)(2)(v).

for Gary F. Spedl
Station Manager
Dresden Station

GFS/JK/maf

Enclosure

cc: J. Martin, Regional Administrator, Region III
NRC Resident Inspector's Office
File/NRC
File/Numerical

210052

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IE22

LICENSEE EVENT REPORT (LER)

Form Rev 2.0

Facility Name (1)	Docket Number (2)	Page (3)
Dresden Nuclear Power Station, Unit 2	0 5 0 0 0 2 3 7	1 of 0 3

Title (4)

High Pressure Coolant Injection [BJ] Minimum Flow Valve Breaker Trip

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 2	1 7	9 4	9 4	— 0 0 3	— 0 0	0 3	1 7	9 4	N/A	
									N/A	

OPERATING MODE (9)	N	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIRMENTS OF 10CFR (Check one or more of the following) (11)									
POWER LEVEL (10)	0 9 9		20.402(b)		20.405(c)		50.73(a)(2)(iv)		73.71(b)		
			20.405(a)(1)(i)		50.36(c)(1)	X	50.73(a)(2)(v)		73.71(c)		
			20.405(a)(1)(ii)		50.36(c)(2)		50.73(a)(2)(vii)		Other (Specify in Abstract below and in Text)		
			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)				

LICENSE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER
Mark A. Churilla, System Engineering Department Ext. 2788	AREA CODE 8 1 5 9 4 2 - 2 9 2 0

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 J	B L K	M 0 8 0	No					

SUPPLEMENTAL REPORT EXPECTED (14)		Expected Submission Date (15)	Month	Day	Year
	Yes (If yes, complete EXPECTED SUBMISSION DATE)	X NO			

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

On February 17, 1994, at 1245 with Unit 2 at 99% rated core thermal power, the Unit 2 High Pressure Coolant Injection (HPCI) Motor Operated (MO) minimum flow valve 2-2301-14 breaker tripped during performance of Special Procedure (SP) 93-12-119, "Generic Letter 89-10 dP Testing". The tripped breaker resulted in the minimum flow valve being inoperable. The Unit 2 HPCI System was already Out-of-Service (OOS) for the performance of the special procedure. The breaker was replaced and the HPCI system was returned to service on February 20, 1994, at 0041 hours. The cause of the trip was determined to be failure of the terminal block due to a loose lug. The breaker was inspected and replaced. Other breakers on the bus were inspected for loose lugs and none were found.

PLANT AND SYSTEM IDENTIFICATION:

General Electric-Boiling Water Reactor-2527 MWt rated core thermal power.

Nuclear Tracking System (NTS) tracking code numbers are identified in the text as (XXX-XXX-XX-XXXXX)

EVENT IDENTIFICATION:

High Pressure Coolant Injection [BJ] Minimum Flow Valve Breaker Trip

A. CONDITIONS PRIOR TO EVENT:

Unit: 2 Event Date: 2/17/94 Event Time: 1245

Reactor Mode: N Mode Name: Run Power Level: 99%

Reactor Coolant System (RCS) Pressure: 996 psi

B. DESCRIPTION OF EVENT:

On February 17, 1994, at 1245, with Unit 2 at 99% rated core thermal power, the Unit 2 High Pressure Coolant Injection (HPCI) Motor Operated (MO) minimum flow valve 2-2301-14 breaker, 250 VDC Bus 2B, tripped while performing Special Procedure SP 93-12-119. The procedure was obtaining data in support of the requirements of Generic Letter 89-10 "Safety-Related Motor-Operated Valve Testing and Surveillance". The system had been placed in a planned seven day Limiting Condition for Operation (LCO) on February 16, 1994. The procedure required MO 2-2301-14 to be cycled several times. The valve breaker failed at the conclusion of the procedure when the HPCI system was being placed back into a standby condition. The valve remained in the open position due to the breaker failure. Initial investigation of the breaker showed that it had sustained significant damage and the valve was subsequently manually closed. Nuclear Work Request (NWR) 24259 was written to investigate and repair the problem. A broken terminal block was found inside the breaker when it was inspected. It is believed that the broken terminal block resulted in shorting of 250 VDC in the breaker. The breaker was replaced with a newer model and tested satisfactorily. The seven day LCO was terminated on February 20, 1994, at 0041 hours.

C. APPARENT CAUSE OF EVENT:

This report is being submitted in accordance with 10CFR50.73(a)(2)(v)(D), which requires the reporting of any condition that could have prevented the fulfillment of a safety system needed to mitigate the consequences of an accident.

It is believed that a short at the breaker was caused by either a loose connection that caused a flash over or previous damage to insulators between two points on the terminal block incurred during installation of the lugs. Performance of this procedure required that this valve be stroked several times to obtain the data needed. Although this stroking was determined to be within the design limits of the valve, it is postulated that it contributed to the conditions leading up to the failure.

D. SAFETY ANALYSIS OF EVENT:

HPCI valve MO 2-2301-14 is used as a minimum flow valve for the HPCI system during operation of the system. Failure of the valve during system operation would cause a loss of minimum flow protection for the pump.

HPCI valve 2-2301-14 and the check valve downstream of it, 2-2301-40, are designated primary containment isolation valves. Due to the breaker trip, the valve failed in the open position. The safety significance is minimal, however, since the check valve, 2-2301-40, was still available to perform a primary containment isolation function and MO 2-2301-14 was manually shut.

In this event the HPCI system was inoperable for the performance of Special Procedure SP 93-12-119. All other Emergency Core Cooling Systems (ECCS) required by TS 3.5.C.2.a were operable during this event. Therefore, the safety significance of this event is considered minimal.

E. CORRECTIVE ACTIONS:

Immediate corrective action after this event included manually isolating HPCI valve, 2-2301-14. This valve and the check valve downstream of it, 2-2301-40, are primary containment isolation valves.

The initial corrective actions replaced the damaged breaker with a new one. Due to the existing breaker being obsolete, a different breaker was installed that required a design change. No corrective actions were determined to be required for the terminal block.

F. PREVIOUS OCCURRENCES:

None

G. COMPONENT FAILURE DATA:

Manufacturer	Nomenclature	Model Number	Mfg. Part Number
Marathon Electric Manufacturing Corporation	BLK	Not Available	Not Available