Commonwealth Edison
Dresden Nuclear Power Station
R.R. #1
Morris, Illinois 60450
Telephone 815/942-2920

February 16, 1993

CWS PMLTR #: 93-0087

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

Licensee Event Report #93-002-0, Docket #050237 is being submitted as required by Technical Specification 6.6, NUREG 1022 and 10 CFR 50.73(a)(2)(i)(A).

for 2-16-93
Charles W. Schroeder
Station Manager
Dresden Station

CWS/slb

Enclosure

cc: A. Bert Davis, Regional Administrator, Region III
File/NRC
File/Numerical

(g:\wpt\cwslt93\93-0087)

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) Type B and C Primary Containment Local Leak Rate Testing Limit Exceeded Due To Leakage Past Head Cooling Inlet Isolation Valve 2-205-2-4

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)
									N/A	0 5 0 0 0 - -
0 1	2 1	9 3	9 3	0 0 2	0 0	0 2	2 0	9 3	N/A	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)			
N		20.402(b)	20.405(c)	50.73(a)(2)(iv)	73.71(b)
POWER LEVEL (10)		20.405(a)(1)(i)	50.36(c)(1)	50.73(a)(2)(v)	73.71(c)
0 0 0		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)	X 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
Michael Andjelic, LLRT Coordinator Ext. 2366	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 P	I S V	C 6 6 5	Y					

SUPPLEMENTAL REPORT EXPECTED (14)		Expected Submission Date (15)	Month	Day	Year
X	Yes (If yes, complete EXPECTED SUBMISSION DATE)		0 9	1 7	9 3
	NO				

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

On January 21, 1993 with Unit 2 in a refuel outage, the performance of Dresden Technical Surveillance (DTS) 1600-01, Local Leak Rate Testing Of Primary Containment Isolation Valves, identified the Head Cooling Inlet Isolation Valve 2-205-2-4 to be leaking an undetermined amount. This leakage rate exceeded the maximum pathway leakage rate for Type B and C primary containment leakage, 488.452 scfh ($0.6L_a$), as listed in Technical Specification 3.7.A.2.b.(2)(a). Once the leakage rate was recorded, the valve was again verified to be in the fully closed position. The measured leakage rate dropped to 3.0 scfh upon increasing the seating force. Work Request (WR) 15642 was written to investigate and repair the valve in order to reduce leakage. The safety significance of the leakage past valve 2-205-2-4 has been considered to be minimal since the redundant Head Cooling Isolation Valve 2-205-27 leaked 3.31 scfh; therefore, the total through leakage out of the penetration, on a minimum pathway basis, was 3.31 scfh and would not cause the maximum off site dose rates established in 10 CFR 100 to be exceeded. The cause of the unsatisfactory leakage past the 2-205-2-4 valve is still under investigation. This valve will be repaired and retested in accordance with DTS 1600-01 prior to the end of the D2R13 refuel outage. A supplement to this report will be submitted by September 17, 1993 to outline the cause of the event, maintenance history, corrective actions, retest results, and component failure data for this valve and any other valves which exceed Station guidelines (237-180-93-00201).

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:

Type B and C Primary Containment Local Leak Rate Testing Limit Exceeded Due To Leakage Past Head Cooling Inlet Isolation Valve 2-205-2-4

A. CONDITIONS PRIOR TO EVENT:

Unit: 2 Event Date: January 21, 1993 Event Time: 0000 hrs

Reactor Mode: N Mode Name: Refuel Power Level: 0%

B. DESCRIPTION OF EVENT:

On January 21, 1993 with Unit 2 in a refuel outage, the performance of Dresden Technical Surveillance (DTS) 1600-01, Local Leak Rate Testing Of Primary Containment Isolation Valves, identified the Head Cooling Inlet Isolation Valve 2-205-2-4 to be leaking an undetermined amount. This leakage rate exceeded the maximum pathway leakage rate for Type B and C primary containment leakage, 488.452 scfh ($0.6L_a$), as listed in Technical Specification 3.7.A.2.b.(2)(a). Once the leakage rate was recorded, the valve was again verified to be in the fully closed position. The actuator was manually engaged and closed with increased force. The measured leakage rate dropped to 3.0 scfh upon increasing the seating force.

The Shift Engineer was notified that the leakage past the Head Cooling Inlet Isolation Valve 2-205-2-4 had caused the total as found Type B and C primary containment leakage rate to exceed $0.6L_a$ (488.452 scfh). A Problem Identification Form (PIF) was initiated per Dresden Administrative Procedure (DAP) 02-27, Integrated Reporting Process. Work Request (WR) 15642 was written to investigate and repair the valve in order to reduce leakage.

C. APPARENT CAUSE OF EVENT:

This report is being submitted in accordance with 10 CFR 50.73(a)(2)(i) which requires the reporting of any operation or condition prohibited by the Technical Specifications.

The cause of the unsatisfactory leakage past the 2-205-2-4 valve is still under investigation. This valve will be repaired and retested in accordance with DTS 1600-01 prior to the end of the D2R13 refuel outage. A supplement to this report will be submitted by September 17, 1993 to outline the cause of the event, maintenance history, corrective actions, retest results, and component failure data for this valve and any other valves which exceed Station guidelines (237-180-93-00201).

D. SAFETY ANALYSIS OF EVENT:

The safety significance of the leakage past valve 2-205-2-4 has been considered to be minimal since the redundant Head Cooling Isolation Valve 2-205-27 leaked 3.31 scfh; therefore, the total through leakage out of the penetration, on a minimum pathway basis, was 3.31 scfh and would not cause the maximum off site dose rates established in 10 CFR 100 to be exceeded.

E. CORRECTIVE ACTIONS:

The actuator for the Head Cooling Inlet Isolation Valve 2-205-2-4 will be diagnostically tested, repaired, and retested in accordance with DTS 1600-01 prior to the end of the D2R13 refuel outage. A supplement to this report will be submitted by September 17, 1993 to outline the cause of the event, maintenance history, retest results, corrective actions, and component failure data for this valve and any other valves which exceed Station guidelines (237-180-93-00201).

F. PREVIOUS OCCURRENCES:

LER/Docket Numbers	Title
90-009/0500237	Type B and C Primary Containment Local Leak Rate Test Requirements Exceeded Due To Leaking Isolation Valves

G. COMPONENT FAILURE DATA:

Manufacturer	Nomenclature	Model Number	Mfg. Part Number

An industry - wide data base search will be performed and included in the supplemental report (237-180-93-00201)