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

April 07, 1993

CWS PMLTR 93-0145

U. S. Nuclear Regulatory Commission
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Licensee Event Report 93-006, Docket 050249 is being submitted as required by Technical Specification 6.6, NUREG 1022 and 10 CFR 50.73(a)(2)(i)(b).

for 4-12-93
Charles W. Schroeder
Station Manager
Dresden Station

CWS/slb

cc: A. Bert Davis, Regional Administrator, Region III
NRC Resident Inspector's Office
File/NRC
File/Numerical

160019

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LICENSEE EVENT REPORT (LER)

Form Rev 2.0

Facility Name (1)	Docket Number (2)	Page (3)
Dresden Nuclear Power Station, Unit 3	0 5 0 0 0 2 4 9	1 of 0 3

Title (4)

Type B and C Primary Containment Local Leak Rate Testing Limit of 0.6La Exceeded due to Leakage Past Inboard Feedwater Check Valve 3-220-58A

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 3	1 3	9 3	9 3	— 0 0 6 —	0 0	0 4	1 2	9 3	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 0 0		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)		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)		

LICENSE CONTACT FOR THIS LER (12)

NAME		TELEPHONE NUMBER	
		AREA CODE	
M. Andjelic, Local Leak Rate 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	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS
X	S J	I S V	G 0 8 0	Yes					

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

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

On March 13, 1993 with Unit 3 in a forced maintenance outage, the performance of Dresden Technical Surveillance (DTS) 1600-01, Local Leak Rate Testing Of Primary Containment Isolation Valves, identified the Inboard "A" Feedwater Line Check Valve 3-220-58A to be leaking 196.67 scfh. This value, when added to the existing maximum pathway 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). The cause of the unsatisfactory leakage past the inboard "A" Feedwater Line Check Valve 3-220-58A is still under investigation. This valve will be inspected and repaired under WR 16938. The safety significance of the leakage past the Inboard "A" Feedwater Line Check Valve 3-220-58A has been considered to be minimal since the redundant Outboard Feedwater Check Valve 3-220-62A leaked 3.65 scfh; therefore, the total through leakage out of the penetration, on a minimum pathway basis, was 3.65 scfh and would not cause the maximum off site dose rates established in 10 CFR 100 to be exceeded. A final as-left local leak rate test will be performed in accordance with DTS 1600-01 to verify the valve's seating integrity prior to placing it back into service. A supplement to this report will be submitted by October 1, 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 during the D3F15 maintenance outage (249-180-93-00601).

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

Form Rev. 2.0

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			Page (3)
		Year	Sequential Number	Revision Number	
Dresden Nuclear Power Station	0 5 0 0 0 2 4 9	9 3	0 0 6	0 0	0 2 OF 0 3

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

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 of 0.6 L_a Exceeded Due To Leakage Past Inboard Feedwater Check Valve 3-220-58A.

A. CONDITIONS PRIOR TO EVENT:

Unit: 3 Event Date: March 13, 1993 Event Time: 1500 hrs

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

B. DESCRIPTION OF EVENT:

On March 13, 1993 with Unit 3 in a forced maintenance outage, the performance of Dresden Technical Surveillance (DTS) 1600-01, Local Leak Rate Testing Of Primary Containment Isolation Valves, identified the Inboard "A" Feedwater Line Check Valve 3-220-58A to be leaking 196.67 scfh. This value, when added to the existing maximum pathway 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).

Upon identification of the failure, the leakage rate was recorded and the Shift Engineer was notified that the leakage past the Inboard "A" Feedwater Line check valve 3-220-58A caused the total measured 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) 16938 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 inboard "A" Feedwater Line Check Valve 3-220-58A is still under investigation. This valve will be repaired under WR 16938 and retested in accordance with DTS 1600-01 prior to the end of the D3F15 maintenance outage. A supplement to this report will be submitted by October 1, 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 during the D3F15 maintenance outage (249-180-93-00601).

D. SAFETY ANALYSIS OF EVENT:

The safety significance of the leakage past the Inboard "A" Feedwater Line Check Valve 3-220-58A has been considered to be minimal since the redundant Outboard Feedwater Check Valve 3-220-62A leaked 3.65 scfh; therefore, the total through leakage out of the penetration, on a minimum pathway basis, was 3.65 scfh and would not cause the maximum off site dose rates established in 10 CFR 100 to be exceeded.

E. CORRECTIVE ACTIONS:

The failure of the Inboard "A" Feedwater Line Check Valve 3-220-58A is still under investigation. This valve will be inspected and repaired under WR 16938. A final as-left local leak rate test will be performed in accordance with DTS 1600-01 to verify its integrity prior to placing this valve back into service. A supplement to this report will be submitted by October 1, 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 during the D3F15 maintenance outage (249-180-93-00601).

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
89-009/0500249	Local Leak Rate Testing "As Found" Limit Exceeded Due to Leakage From Primary Containment Valves

G. COMPONENT FAILURE DATA:

Manufacturer	Nomenclature	Model Number	Mfg. Part Number
Crane	Feedwater Check Valve 3-220-58A	973	N/A

An industry - wide data base search will be performed and included in the supplemental report (249-180-93-00601).