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

January 2, 1990

EDE LTR #89-959

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

Licensee Event Report #89-009-0, Docket #050249 is being submitted as required by Technical Specification 6.6, NUREG 1022 and 10 CFR 50.73(a)(2)(i)(B).

E. D. Eenigenburg
Station Manager
Dresden Nuclear Power Station

EDE/jmt

Enclosure

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

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1/1

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)

Local Leak Rate Testing "As Found" Limit Exceeded Due to Excessive Leakage From Primary Containment Valves

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
1 2	0 7	8 9	8 9	0 0 9	0 0	0 1	0 2	9 0	N/A	0 5 0 0 0

OPERATING MODE (9): N

POWER LEVEL (10): 0 0 0

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)	___	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	
M. Horbaczewski, Technical Staff Group Leader Ext. 2461	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	L K	X X X V	H 0 3 7	Y					

SUPPLEMENTAL REPORT EXPECTED (14)

X Yes (If yes, complete EXPECTED SUBMISSION DATE) ___ NO

Expected Submission Date (15): Month 0 4 | Day 0 1 | Year 9 0

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

On December 7, 1989, with Unit 3 in a refueling outage and during the performance of Dresden Technical Staff Surveillance Procedure (DTS) 1600-1, Local Leak Rate Testing (LLRT) of Primary Containment Isolation Valves, feedwater check valve 3-0220-58A leaked 1062.82 Standard Cubic Feet per Hour (SCFH). This leak rate, along with unsatisfactory leak rates from Reactor Building to pressure suppression chamber vacuum breaker check valve 3-1601-31B and High Pressure Coolant Injection steam exhaust to torus check valve 3-2301-45, combined to bring the total as-found leakage using the maximum pathway method for type 'B' and 'C' testing to 1510.4526 SCFH, which exceeded the Technical Specification 3.7.A.2.b.(2)(a) limit of 488.452 SCFH. The cause of the excessive leakage is unknown at this time. These valves will be repaired and tested prior to unit startup. A supplement to this report will then be submitted outlining the cause of failure, retest results, the final type B and C leak rate test results and any additional corrective actions to prevent recurrence. Also included with the supplement will be a tabulation of all the testable penetrations' maximum and minimum pathway leakages. The safety significance of this event was minimal because in line valves were not observed to be leaking. Therefore, the "through" leakage, which represents actual containment leakage, was minimal. A previous occurrence of this type is outlined in reportable occurrence 88-27 on Docket 050-249.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) Year	Sequential Number	Revision Number	Page (3)
Dresden Nuclear Power Station	0 5 0 0 0 2 4 9	8 9	- 0 0 9	- 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:

Local Leak Rate Testing "As Found" Limit Exceeded Due to Excessive Leakage From Primary Containment Valves [JM]

A. CONDITIONS PRIOR TO EVENT:

Unit: 3 Event Date: December 7, 1989 Event Time: 0900

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

Reactor Coolant System (RCS) Pressure: 0 psig

B. DESCRIPTION OF EVENT:

On December 7, 1989, with Unit 3 in a Refuel Outage, while performing Dresden Technical Staff Surveillance (DTS) 1600-1, Local Leak Rate Testing (LLRT) of Primary Containment Isolation Valves, feedwater [SJ] check valve 3-0220-58A leaked 1062.82 standard cubic feet per hour (SCFH). Previously, LLRTs of High Pressure Coolant Injection (HPCI) [BJ] steam exhaust to pressure suppression chamber check valve 3-2301-45 and reactor building [NG] to pressure suppression chamber vacuum breaker check valve 3-1601-31B [BF] revealed calculated leakages of 100.5 and 159.08 SCFH respectively. Inclusion of these test results into the total "As Left" type B and C LLRT data resulted in a total calculated leakage of 1510.4526 SCFH. This exceeded the Technical Specification 3.7.A.2.b.(2)(a) limit of 488.452 SCFH for total calculated type B and C leakage. These calculations utilize a maximum pathway methodology, which includes all test volume leakage even though redundant isolation valves were available to prevent actual through leakage. Work Requests 86069, 89030, and 89003 were initiated to repair the valves.

C. APPARENT CAUSE OF EVENT:

The cause of the unsatisfactory leakages from the HPCI steam exhaust to torus check valve 3-2301-45, Reactor Building to pressure suppression chamber vacuum breaker check valve 3-1601-31B and feedwater check valve 3-0220-58A is under investigation. These valves will be repaired and retested per DTS 1600-1 prior to unit startup. A supplement to this report will then be submitted to report the cause of failure, the retest results, and the final type "B" and "C" leak rate test results. This report is being submitted in accordance with 10CFR 50.73(a)(2)(i)(B), which requires the reporting of any operation or condition prohibited by the Technical Specifications.

D. SAFETY ANALYSIS OF EVENT:

The safety significance has been considered minimal because in line HPCI stop check valve 3-2301-74, in line isolation valve (AO3-1601-20B) for reactor building to pressure suppression chamber vacuum breaker check valve 2-1601-31B, and in line feedwater check valve 3-0220-62A were not observed to be leaking. Therefore, the "through" leakage for these primary containment penetrations was minimal.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) Year	Sequential Number	Revision Number	Page (3)
Dresden Nuclear Power Station	0 5 0 0 0 2 4 9	8 9	- 0 0 9 -	0 0	0 3 OF 0 3

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

E. CORRECTIVE ACTIONS:

Prior to unit startup, the cause of the excessive leakage from the 3-0220-58A, 3-1601-31B and 3-2301-45 valves will be determined and appropriate repairs will be completed by Mechanical Maintenance personnel (249-200-89-11801). The valves will be retested by the Technical Staff per DTS 1600-1. A supplement to this investigation report outlining the cause of the event, the retest results, the final type "B" and "C" leak rate test results, and any additional corrective actions deemed necessary will also be submitted. In addition, a tabulation of all the testable penetrations maximum and minimum pathway leakages will be included (249-200-89-11802).

F. PREVIOUS OCCURRENCES:

LER/Docket Numbers	Title
89-4/050249	Type B and C Local Leak Rate Test Limit Exceeded Due to Leakage Through Primary Containment Isolation Valve

A major contributor to this event was leakage from containment atmosphere dilution [BB] valve 3-2599-23B (redundant isolation valve not leaking). The cause of this event was attributed to a small amount of foreign material deposited on the valve seating surface. This foreign material also caused the valve seating surface to become slightly worn.

G. COMPONENT FAILURE DATA:

This information will be included in a supplemental report.