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

September 11, 1992

CWS LTR #92-421

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

Attached please find Licensee Event Report #89-009-1, Docket #050249. This revised report is being submitted to provide an update of corrective actions implemented as stated in the previous revision of the report and further clarifying information.

L. F. Gerner for
C. W. Schroeder
Station Manager
Dresden Nuclear Power Station

Enclosure

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

ZDVR/682

IE22
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 6

Title (4)

Local Leak Rate Testing "As Found" Limit Exceeded Due to 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	
1 2	0 7	8 9	8 9	0 0 9	0 1	0 1	0 2	9 0	N/A	

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)	0 0 0	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
Michael S. Andjelic, Technical Staff Engineer 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 J	X I S V	M 3 6 0	Y	X	J B	X I S V	C 2 5 6	Y
A	B F	X I S V	C 6 8 4	Y					

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 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 exhibited a leakage of 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 pressure suppression chamber 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. This leakage exceeded the Technical Specification 3.7.A.2.b.(2)(a) limit of 488.452 SCFH. 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 LER-88-27 on Docket 050-249.

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 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 exhibited a leakage of 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-Found" 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.

In addition to the above valves, eight other test volumes required repairs and adjustments due to unsatisfactory leakage rates. These test volumes and their corresponding "as-found" leakage rates are listed below:

TEST VOLUME	SYSTEM	"As-Found" Type B&C (Maximum Pathway) LEAKAGE RATE (SCFH)	"As-Found" Type A (Minimum Pathway) LEAKAGE RATE (SCFH)
3-205-2-4	REACTOR HEAD COOLING	62.71	11.48
3-220-1,2	MAIN STEAM DRAIN	52.05	6.94
3-220-57B, 58B	FEEDWATER	1348.6	23.27
3-1001-1A, 1B, 2A, 2B, 2C	SHUTDOWN COOLING	107.54	53.77

TEST VOLUME	SYSTEM	"As Found" Type B&C (Maximum Pathway) LEAKAGE RATE (SCFH)	"As Found" Type A (Minimum Pathway) LEAKAGE RATE (SCFH)
3-1601-57, 58, & 59	NITROGEN MAKEUP	74.53	3.26
3-1601-21, 22, 55, 56, & 8502-500	NITROGEN INERTING	65.31	32.65
3-3703, 3706	REACTOR BUILDING CLOSED COOLING WATER (RBCCW)	62.39	9.84
3-8501-5A	DRYWELL AIR SAMPLE	33.48	23.28

C. APPARENT CAUSE OF EVENT:

The HPCI steam exhaust to pressure suppression chamber check valve 3-2301-45 was disassembled and inspected under Work Request (WR) 83971. An inspection of the valve internals revealed that the valve seats were worn and cracked. This degradation of the valve seats was primarily due to service conditions (saturated steam) and valve age.

Soap-bubble testing of the reactor building to pressure suppression chamber vacuum breaker check valve 3-1601-31B revealed leakage past the outer shaft flange and the valve disk-to-seat surface. In addition, a visual inspection was performed through an access hole in the piping upstream of the valve. This inspection revealed a layer of grease on the backside, non-seating side, of the valve disk along the edge of the valve seat; however, the valve seating surfaces were free of grease. Leakage past the outer shaft seal flange was due to the valve shaft flange being installed incorrectly. The flange was previously installed with its raised face outward and not towards the flange gasket. The sealing capability of the flange was further degraded since it was discovered that the flange tensioning nuts were improperly torqued (loose when turned by hand). Leakage past the valve disk-to-seat region was caused by worn hinge pins which prevented the valve disk to center in its seat. An investigation regarding the presence of grease on the backside of the valve disk determined that the grease was applied by mechanical maintenance personnel prior to 1983. The effect of the grease was judged to be insignificant since the seating surface was free of grease, and the local leak rate test results only increased from 12 to 16 scfh after the grease was removed.

Feedwater check valve 3-220-58A was disassembled and inspected under WR 89069. An examination of the valve internals revealed that there was no damage to the valve disk or seat. It was determined that debris between the seating surfaces of the valve disk and seat was the cause of the excessive leakage.

A summary table describing the root cause and corrective actions for the remaining test volumes which exhibited excessive leakage rates is contained in Section E of this report.

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 (A03-1601-20B) for reactor building to pressure suppression chamber vacuum breaker check valve 3-1601-31B, 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.

In addition, the total as-found minimum pathway leakage rate, which is a more accurate representation of actual potential containment "through" leakage under accident conditions, was calculated to be 339.1 scfh. This total was within the Technical Specification limit of 488 scfh for all type B and C tests.

E. CORRECTIVE ACTIONS:

Due to seating surface degradation, High Pressure Coolant Injection steam exhaust to pressure suppression chamber check valve 3-2301-45 was replaced under WR 83971. To improve seat life, the replacement check valve was equipped with Viton seats versus the Buna seats contained in the original check valve. A post-maintenance local leak rate test was performed on January 11, 1990 and yielded a satisfactory leakage rate of 0.0 scfh.

Leakage from the reactor building to pressure suppression chamber vacuum breaker check valve 3-1601-31B was repaired under WR 89030 and WR 89141. The valve hinge pins were removed, cleaned, inspected, and reinstalled with the hinge pins rotated 180 degrees from their original orientation. Rotating the hinge pins 180 degrees raised the disk-to-seat orientation which allowed the disk to center in its seat. Leakage past the shaft end flange was corrected by reinstalling the flange with a new gasket and torquing the flange tensioning nuts to the appropriate torque value. A final local leak rate test was performed on February 5, 1990. The results of this test indicated a satisfactory leakage rate of 13.87 scfh. Since the valve shaft flange was installed incorrectly during the performance of DOS 1600-13, "Quarterly Suppression Chamber to Reactor Building Vacuum Breaker Full Stroke Test of 2(3)-1601-31A and B", this procedure was revised to allow the use of existing access ports to demonstrate valve operability. This eliminates the need to remove the shaft flanges to perform the quarterly surveillance.

Feedwater check valve 3-220-58A was repaired under WR 89069. The valve was disassembled and all sealing surfaces were cleaned and inspected. The seal ring and a dirty O-ring were replaced. A post-maintenance local leak rate test was performed on January 16, 1990. This test resulted in a satisfactory leakage rate of 3.79 scfh.

A table is provided below to summarize the repairs and adjustments performed on the remaining test volumes which had exhibited unsatisfactory leakage rates.

Test Volume:	Repairs and Adjustments
3-205-2-4	Reactor head cooling valve 3-205-2-4 was disassembled and inspected per Work Request 89865. The inspection revealed cuts on the seating surface of the disk. The disk/seat assembly was lapped and the valve was repacked. A final local leak rate test yielded a satisfactory leakage rate of 11.7 scfh. Maintenance records dating back to 1986 indicate that no previous mechanical failures relating to the disk/seat assembly have occurred. In addition, LLRT records dating back to 1985 indicate no previous failures of this valve have occurred.
3-220-1,2	Main Steam drain valve 3-220-1 was disassembled and inspected per Work Request D-89072. The inspection revealed that the stem/disk assembly was worn but the body seat was in fair condition. A new stem/disk assembly was installed and the body seat was lapped for a full 360 degree contact. The valve was repacked and a local leak rate test was performed. The local leak rate test yielded a satisfactory leakage rate of 6.94 scfh. Maintenance records dating back to 1986 revealed that there were no previous failures relating to the valve seating surfaces. In addition, LLRT records dating back to 1985 indicate no previous failures of this valve have occurred.

Test Volume:	Repairs and Adjustments
3-220-57B, 58B	Feedwater check valve 3-220-58B was disassembled and inspected under Work Request D-89370. The valve disk and seat were inspected and found to be in good condition. The disk and seat were then polished and both the O ring and the seal ring were replaced. An as-left local leak rate test was performed and the results yielded a satisfactory leakage rate of 13.33 scfh. Maintenance history indicates no similar valve failures have been reported. In addition, LLRT records dating back to 1985 indicate no previous failures of this valve have occurred.
3-1001-1A, 1B, 2A, 2B, & 2C	WR 89401 was written to repair shutdown cooling valve 3-1001-1A The valve was subsequently repaired during the D3R12 refueling outage. The as-left leakage rate for the volume was 0.29 scfh.
3-1601-57, 58, 59	This test volume failed its local leak rate test due to nitrogen relief valve 3-8526. The valve was inspected under Work Request 89032. The relief valve was removed and placed on a test stand. Once on the test stand, it was determined that the valve had a missing blow ring set screw. Consequently, the valve was replaced with a new unit and a final local leak rate test was performed. This test resulted in a satisfactory leakage rate of 3.26 scfh. Maintenance history indicates that the valve seat was cleaned in 1985. During the maintenance of this valve, teflon tape was found on the valve seat. LLRT records dating back to 1985 indicate this to be the only failure related to this valve.
3-1601-21, 22, 55, 56, & 8502-500	Valve 3-1601-21 was inspected and repaired under Work Request 89031. A close inspection of the valve revealed that a stroke adjustment was required. The valve stroke was then adjusted to the required setting. In addition, Work Request 90315 was initiated to inspect and repair valve 3-1601-55. The solenoid valve body was replaced and a final local leak rate test was performed. The final leakage rate for the test volume was a satisfactory 27.31 scfh. Maintenance records indicate that the 3-1602-21 valve was replaced in 1986 and in 1988. The valve air operator was also rebuilt in 1988. No previous failures have been associated with improper adjustments.
3-3703, 3706	Work Request 89371 was initiated to repair excessive leakage past motor operated valve 3-3703. This leakage was corrected by performing a torque switch adjustment to increase the valve thrust. A post-maintenance local leak rate test was performed and a satisfactory leakage rate of 9.84 scfh was obtained. Maintenance history indicates that the motor operator was rebuilt in 1988 and that the valve was repacked in 1985. LLRT records dating back to 1985 indicate no other failures.
3-8501-5A	Drywell air sample valve 3-8501-5A was inspected and repaired under Work Request 89821. Upon disassembly and inspection of the valve internals, excessive amounts of dirt and grit were found in the seating area of the disk and seat. The valve was cleaned, reassembled, and repacked. A post-maintenance local leak rate test was performed and it yielded a satisfactory leakage rate of 0.0 scfh. Maintenance history indicates that no previous valve failures of this type have been reported. In addition, LLRT records dating back to 1985 indicate no previous failures have occurred.

After all repairs and adjustments were made, the total as-left maximum pathway leakage rate was calculated to be 485.43 scfh. The total minimum pathway leakage rate, which is a more accurate representation of actual potential containment "through" leakage under accident conditions, was calculated to be 241.17 scfh. Both of these totals are within the Technical Specification limit of 488 scfh for all type B and C tests.

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 was leakage from containment atmosphere dilution [BB] valve 3-2599-23B (redundant isolation valve not leaking). The cause 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:

Manufacturer	Nomenclature	Model Number	Mfg. Part Number
Crane Co.	Reactor Building to Pressure Suppression Chamber Vacuum Breaker 3-1601-31B	L123A	N/A
Mission Drilling Prod.	HPCI Turbine Exhaust 3-2301-45	DuoCheck	N/A
Crane Co.	Inboard Feedwater Check Valve 3-220-58A	973	N/A

An industry wide NPRDS data base search revealed that 174 records of failures for the Mission DuoCheck check valve. Causes for the failures have included worn seats, wrong seat application, erosion, and worn/broken hinge and stop pins.

An NPRDS search of the Crane Co. Model 973 check valve revealed 85 reported failures, which included missing hold down bolts, leaking bonnets, loose hinge pins, degraded O-rings, corrosion, hinge pin wear, and dirty and galled seats.

The Crane Co. Model L123A check valve had one NPRDS failure report of bonnet-to-body leakage.