Commonwealth Edison
Dresden Nuclear Power Station
6500 North Dresden Road
Morris, Illinois 60450
Telephone 815/942-2920

March 30, 1994

GFSLTR 94-102

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

Licensee Event Report 93-016-01, Docket 50-249 is being submitted as required by Technical Specification 6.6, NUREG 1022 and 10CFR50.73(a)(2)(i)(B).

Additionally, this LER is being submitted to report the exceeding of an administrative limit for Primary Containment Leakage. This administrative limit was established as a condition of being granted a scheduler exemption from the local leak rate testing surveillance interval required by 10CFR50, Appendix J. Two ENS phone notifications were also performed concerning these events.

Sincerely,

for 4-1-94

Gary F. Spedl
Station Manager
Dresden Station

GFS/MMcG/cfq

enclosure

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

IE22

NRC FORM 366 (5-92) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 5/31/95

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Dresden 3	05000249	1 OF 7

TITLE (4)
Type B and C Test Leakage Limit of 0.6.L_a Exceeded Due to Leaking Valve.

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 NAME	DOCKET NUMBER
09	29	93	93	-- 016 --	01	10	29	93	None	
									FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9): N
POWER LEVEL (10): 099

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (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
	20.405(a)(1)(iii)	X	50.73(a)(2)(i)		50.73(a)(2)(viii)(A)		(Specify in Abstract below and in Text, NRC Form 366A)
	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 (Include Area Code)
M. McGivern, Local Leak Rate Test Coordinator Ext. 2526	(815) 942-2920

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	BB	ISV	H037	Yes	X	BF	ISV	C665	Yes
X	IK	ISV	R340	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 15 single-spaced typewritten lines) (16)

At approximately 1600, on September 29, 1993 with Unit 3 operating at 99% power, the performance of Dresden Special Procedure SP 93-8-73, Local Leak Rate Testing Of Primary Containment Isolation Valves During Reactor Operation, identified the Atmospheric Containment Atmosphere Dilution (ACAD) System [BB] Check Valve 3-2599-23A to be leaking an undetermined amount. 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 safety significance of the leakage past valve 3-2599-23A was considered to be minimal since the additional leakage out of containment, on a minimum pathway basis, was 7.93 scfh from the inboard isolation valve 3-2599-2A and would not cause the maximum off-site dose rates established in 10 CFR 100 to be exceeded. Upon completion of maintenance, the valve was retested. This test yielded a leakage rate of 2.44 scfh. This supplement has been submitted to report the cause and corrective actions for three other valve failures which occurred during the course of the Unit 3 exemption period.

A. PLANT CONDITIONS PRIOR TO EVENT:

Unit: 3 Event Date: September 29, 1993 Event Time: 1600 hrs

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

Reactor Coolant System Pressure: 1003 psig

B. DESCRIPTION OF EVENT:

ACAD System Check Valve 3-2599-23A

At approximately 1600, on September 29, 1993 with Unit 3 operating at 99% power, the performance of Dresden Special Procedure SP 93-8-73 revision 0, Local Leak Rate Testing Of Primary Containment Isolation Valves During Reactor Operation, identified the Atmospheric Containment Atmosphere Dilution (ACAD) System [BB] Check Valve 3-2599-23A to be leaking an undetermined amount. 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).

In order to verify Primary Containment could still be maintained, the inboard ACAD System isolation valve 3-2599-2A was challenged with a local leak rate test (LLRT), which yielded a leakage rate of 7.93 scfh. The new sum of this pathway's maximum pathway leakage rate, when added to the current sum of Type B and C leakage, resulted in the new total leakage rate being 288.70 scfh. The Shift Control Room Engineer (SCRE) was notified of the event and that with valve 3-2599-2A closed, primary containment integrity could be maintained and reactor power operation could continue per Technical Specification 3.7.D.2. Valve 3-2599-23A was declared inoperable and valves 3-2599-2A and 3-2599-6A were taken Out of Service in the closed position. This ensured that Primary Containment could be maintained by preventing the inadvertent opening of the inboard isolation valve. Additionally, since the leakage rate exceeded the administrative leakage limit of 80% of $0.6L_a$ (390.76 scfh) which was established as a condition of being granted a schedular exemption (by NRR) from the testing interval required by 10 CFR 50, Appendix J, a courtesy ENS phone notification was made at 1805 on September 29, 1993.

A Problem Identification Form (PIF) was initiated per Dresden Administrative Procedure (DAP) 02-27, Integrated Reporting Process. Check Valve 3-2599-23A was repaired under Work Request (WR) D20130, which had previously been submitted on June 16, 1993 as a contingency work request pending as-found LLRT results for the upcoming Unit 3 Refueling Outage D3R13.

Maintenance was completed on October 1, 1993 after which the valve was retested. This test yielded an acceptable leakage rate of 2.44 scfh. The ACAD System check valve was declared operable and the system returned to service.

H_2/O_2 Monitor System Check Valve 3-2499-28A

At approximately 1015, on February 18, 1994 with Unit 3 operating at 66 % power, the performance of Dresden Special Procedure SP 93-8-73 identified the 3A post-accident Containment Atmosphere Hydrogen/Oxygen Monitor System [IK] Check Valve 3-2499-28A to be leaking 76.74 scfh, a value in excess of the valve's administrative leakage limit of 10 scfh. This value when added to the existing maximum pathway leakage rate resulted in the maximum pathway leakage rate for Type B and C primary containment leakage being 350.27 scfh, which is less than the Technical Specification limit of 488.452 scfh ($0.6L_a$).

In order to verify Primary Containment could still be maintained, the inboard H_2/O_2 Monitor System isolation valve 3-2499-29A was challenged with an LLRT, which yielded a leakage rate of 0 scfh. The new sum of this pathway's maximum pathway leakage rate, when added to the current sum of Type B and C leakage, resulted in the new total leakage rate being 273.53 scfh. The Shift Control Room Engineer (SCRE) was notified of the event and that with valve 3-2499-29A closed, primary containment integrity could be maintained and reactor power operation could continue per Technical Specification 3.7.D.2. Valve 3-2499-28A was declared inoperable and valves 3-2499-29A and 3-2499-27A were taken Out of Service in the closed position. This ensured that Primary Containment could be maintained by preventing the inadvertent opening of the inboard isolation valve.

A Problem Identification Form (PIF) was initiated per Dresden Administrative Procedure (DAP) 02-27, Integrated Reporting Process. Check Valve 3-2499-28A was repaired under Work Request (WR) D20151. Maintenance was completed on February 25, 1994 after which the valve was retested. This test yielded an acceptable leakage rate of 0.41 scfh. The 3A H_2/O_2 Monitor System check valve was declared operable and the system returned to service.

H_2/O_2 Monitor System Check Valve 3-2499-28B

At approximately 1300, on February 26, 1994 with Unit 3 operating at 64% power, the performance of Dresden Special Procedure SP 93-8-73 identified the 3B post-accident Containment Atmosphere Hydrogen/Oxygen Monitor System [IK] Check Valve 3-2499-28B to be leaking an undetermined amount. 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).

In order to verify Primary Containment could still be maintained, the inboard H_2/O_2 Monitor System isolation valve 3-2499-29B was challenged with an LLRT, which yielded a leakage rate of 0 scfh. The new sum of this pathway's maximum pathway leakage rate, when added to the current sum of Type B and C leakage, resulted in the new total leakage rate being 271.91 scfh. The Shift Control Room Engineer (SCRE) was notified of the event and that with valve 3-2499-29B closed, primary containment integrity could be maintained and reactor power operation could continue per Technical Specification 3.7.D.2. Valve 3-2499-28B was declared inoperable and valves 3-2499-29B and 3-2499-27B were taken Out of Service in the closed position. This ensured that Primary Containment could be maintained by preventing the inadvertent opening of the inboard isolation valve. Additionally, since the leakage rate exceeded the Technical Specification leakage limit of 488.452 scfh ($0.6L_a$), an ENS phone notification was made at 1626 Central Standard Time on February 28, 1994.

A Problem Identification Form (PIF) was initiated per Dresden Administrative Procedure (DAP) 02-27, Integrated Reporting Process. Check Valve 3-2499-28B was repaired under Work Request (WR) D20150. Maintenance was completed on March 4, 1994 after which the valve was retested. This test yielded an acceptable leakage rate of 0.29 scfh. The

3B H_2/O_2 Monitor System check valve was declared operable and the system returned to service.

Torus to Reactor Building Vacuum Breaker 3-1601-31B

At approximately 1135, on March 2, 1994 with Unit 3 operating at 64% power, the performance of Dresden Special Procedure SP 93-8-73 identified the 3B Torus to Reactor Building Vacuum Breaker System [BF] Check Valve 3-1601-31B to be leaking 259.57 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).

In order to verify Primary Containment could still be maintained, the inboard Vacuum Breaker Butterfly Valve 3-1601-20B had to be challenged with an LLRT. In order to do this, the leakage from around O-rings, which seal around the check valve hinge pin shaft, needed to be stopped. The Shift Control Room Engineer (SCRE) was notified of the event. Using the contingency Work Request D12848, the bolting on the flanges was tightened up and leakage dropped to 0.41 scfh and primary containment integrity was restored.

A Problem Identification Form (PIF) was initiated per Dresden Administrative Procedure (DAP) 02-27, Integrated Reporting Process. Since the leakage rate exceeded the Technical Specification leakage limit of 488.452 scfh ($0.6L_a$), an ENS phone notification was made at 1240 Central Standard Time on March 2, 1994.

C. 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 ACAD System lift-type check valve 3-2599-23A was inspected to determine cause of leakage. Both seating surfaces of the piston and the seat in the body were corroded. The seat had a small scratch across the seating area, however, this scratch did not cause the undetermined leakage. Corrosion and debris found on the seating surfaces would not allow the valve to close properly, thus yielding the leakage path. The cause of the corrosion is from moist air originating from the drywell and ACAD Compressor condensing in the line. Neither the check valve nor the attached piping is heat traced. A review of maintenance history since 1985 shows that one other failure occurred (WR D04143), in that the piston was found stuck to its guide and debris was found on the seat. LLRT records dating back to 1980 indicate no other failures of this valve.

The H_2/O_2 Monitor System lift-type check valve 3-2499-28A was disassembled and inspected in order to determine the cause of the leakage. The inspection revealed no corrosion or degraded valve internals. Black debris, thought to be corrosion from the system's stainless steel piping, was found on the lift check piston and valve body seating areas. The cause of the corrosion is from moist air, originating from the drywell, condensing in the line and corroding the pipe interior. Due to this recurring problem, a Site Engineering Services Request (SESR) 94-015 was initiated for evaluation of the following technical resolutions: redesign of the monitor return piping, check valve replacement and heat tracing of the return line piping (249-180-94-00101).

The H_2/O_2 Monitor System lift-type check valve 3-2499-28B was disassembled and inspected in order to determine the cause of the leakage. The inspection revealed a small scratch on the seating surface, however, this scratch did not cause the undetermined leakage. Black debris, thought to be corrosion from the system's stainless steel piping, was found on the lift check piston and valve body seating areas. The cause of the corrosion is from moist air, originating from the drywell, condensing in the line and corroding the pipe interior. Due to this recurring problem, a Site Engineering Services Request (SESR) 94-015 was initiated for evaluation of the following technical resolutions: redesign of the monitor return piping, check valve replacement and heat tracing of the return line piping (249-180-94-00101). For additional information, see LER/Docket Number 94-001/0500249.

The Torus to Reactor Building Vacuum Breaker 3-1601-31B failed its LLRT due to improper post-operability testing. The vacuum breaker, a split body style swing check valve made by Crane/Chapman, has a removable arm which may be used to operate the valve. Prior to manual operation, a flange is removed allowing the removable arm to be inserted into the hinge pin shaft. After cycling the check valve disk, the arm is removed and the flange replaced and tightened down. This flange keeps leakage from around the shaft O-rings from leaking out. This flange is part of Primary Containment. Prior to January 29, 1987, an LLRT was required to be performed following the quarterly vacuum breaker operability surveillance to ensure the flange was on tightly and there was no primary containment leakage. However, in 1987, an On-Site review of a procedure revision to Dresden Operating Surveillance (DOS) 1600-13, Suppression Chamber To Reactor Building Vacuum Breaker Operability Test For 2/3-1601-31A&B, concluded that the hinge pin shaft flanges were not part of Primary Containment and as such no LLRT was required as part of the surveillance. During an investigation into Torus to Reactor Building vacuum breaker LLRTs (February 1990) it was concluded that the 1987 On-Site Review had been incorrect in their determination that the flange was not part of Primary Containment. In addition, DOS 1600-13 did not contain adequate instructions to verify proper installation of the flange. The investigation team recommended that the vacuum breakers access ports, which allow access to the valve disk, should be used in the performance of the operability surveillance and that the corresponding DOS procedure be revised, thus eliminating the need to remove the hinge pin shaft flange (and the subsequent required as-left LLRT). This access port is outside of Primary Containment and allows for cycling of the vacuum breaker valve disk with a broom handle. This procedure was revised on October 22, 1993 but not until Operations Department Personnel had performed the surveillance (removed and replaced the flange, thus altering Primary Containment without an as-left LLRT) 11 times after the recommendation. LLRT records dating back to 1985 indicate two other failures of this valve.

Dresden Unit 3 had been granted a schedular exemption (by NRR) from the testing interval required by 10 CFR 50, Appendix J. The exemption entailed performing LLRTs on those volumes which could be tested with the reactor at power operation and exempts those volumes which would require the reactor to be shutdown. This exemption lasts until the Unit 3 refueling outage D3R13, which commenced March 10, 1994.

D. SAFETY ANALYSIS:

The safety significance of the leakage past valve 3-2599-23A was considered to be minimal since the redundant ACAD System isolation valve leaked 7.93 scfh; therefore, the total leakage out of the penetration, on a minimum pathway basis, was 7.93 scfh. The leakage past the 3-2499-28A and the 3-2499-28B was considered minimal since leakage pressurizes the H_2/O_2 Monitor System closed loop, which is an extension of the Primary Containment during an accident. Leakage from the closed loops were 0.91 scfh and 0.1 scfh for A and B loops respectively. The leakage past the 3-1601-31B was considered minimal since the inboard vacuum breaker butterfly valve 3-1601-20B conservatively leaked 0.41 scfh; therefore, the total leakage out of the penetration, on a minimum pathway basis, was 0.41 scfh. The current as-left leakage (Type A test) is .6706 wt%/day. If the minimum pathway leakage of the 4 failures , .0185 wt%/day (9.35 scfh), is added to the current as-left minimum pathway leakage total (.6706 wt%/day), then the new total leakage would be .6863 wt%/day. This total is less than the Technical Specification limit of $0.75L_a$ (1.2 wt%/day); therefore, the maximum off-site dose rates established in 10 CFR 100 would not be exceeded in the event of a LOCA.

E. CORRECTIVE ACTIONS:

The ACAD System Check Valve 3-2599-23A was inspected and repaired under WR 20130. Based on the results of the inspection, both the piston and the seat were replaced. After the check valve was reassembled, the final as-left leakage rate was 2.44 scfh.

The 3A Containment Atmosphere H_2/O_2 Monitor System return line check valve 3-2499-28A was inspected and repaired under WR D20151. Based on the results of the inspection, the corrosion products were removed and the seating surfaces were cleaned. After the check valve was reassembled, the final as-left leakage rate was 0.41 scfh. Currently, the 3-2499-28A is a lift check valve with metal to metal seats. Due to this design and light weight, this check is very susceptible to leakage caused by debris. SESR 94-015 involves the investigation to find a check valve with a different design. A potential design is a swing check valve with rubber seats which would be less susceptible to leakage caused by debris on the seats. In addition, the SESR will evaluate redesign and heat tracing of the monitor return piping. (NTS 249-180-94-00101) Resolutions are expected to be in place by the end of the present Refuel Outage D3R13.

The 3B Containment Atmosphere H_2/O_2 Monitor System return line check valve 3-2499-28B was inspected and repaired under WR D20150. Based on the results of the inspection, the corrosion products were removed and the seating surfaces were cleaned. After the check valve was reassembled, the final as-left leakage rate was 0.29 scfh. Currently, the 3-2499-28B is a lift check valve with metal to metal seats. Due to this design and light weight, this check is very susceptible to leakage caused by debris. SESR 94-015 involves the investigation to find a check valve with a different design. A potential design is a swing check valve with rubber seats which would be less susceptible to leakage caused by debris on the seats. In addition, the SESR will evaluate redesign and heat tracing of the monitor return piping. (NTS 249-180-94-00101) Resolutions are expected to be in place by the end of the present Refuel Outage D3R13.

The Torus to Reactor Vacuum Breaker 3-1601-31B shaft hinge pin flanges were tightened under WR D12848 and the final as-left leakage rate was 0.41 scfh. The revised operability surveillance procedure allows for stroking the vacuum breaker from the access port vice using the removable arm and possibly altering Primary Containment.

F. PREVIOUS OCCURRENCES:

LER/Docket Number	Title
92-031/0500237	Failure of the Outboard Drywell Air Sample Valve 2-8501-5B During Its 24 Month Local Leak Rate Testing Surveillance Due to Improper Valve Seating
88-004/0500249	Local Leak Rate Test Limit Exceeded Due to Leakage Through Primary Containment Isolation Valves

G. COMPONENT FAILURE DATA:

Manufacturer	Nomenclature	Model Number	Mfg. Part Number
Hancock	ACAD Check Valve 3-2599-23A	5580W	N/A

An industry - wide data base search revealed nineteen failures for the Hancock Model 5580W lift-type check valve. Thirteen failures were attributed to debris and corrosion of valve internals not allowing the valve to close. One of the failures reported was used in a similar system application.

Manufacturer	Nomenclature	Model Number	Mfg. Part Number
Rockwell Int.	H_2/O_2 Check Valve 3-2499-28A 3-2499-28B	N/A	N/A

An industry - wide data base search revealed no failures for the Rockwell International Company lift-type check valve. This 3/4" check valve does not meet the dimension requirements for reportability.

Manufacturer	Nomenclature	Model Number	Mfg. Part Number
Crane/Chapman	Vacuum Breaker 3-1601-31B	L 123A	N/A

An industry - wide data base search revealed eighteen failures for the Crane/Chapman Model L 123A swing check valve. None of the failures were attributed to loose flange bolting.