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Dresden Nuclear Power Station
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May 25, 1993

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Licensee Event Report 92-31-2 Docket No. 050237 is being submitted as required by Technical Specification 6.6, NUREG 1022, and 10 CFR 50.73(a)(2)(i)(B).

This revised report is being submitted to identify the cause and corrective actions for Primary Containment Isolation Valve failures identified during the Dresden Unit 2 10 CFR 50, Appendix J exemption period granted by NRR.

for 5-28-93
Charles W. Schroeder
Station Manager
Dresden Station

CWS/slb

Enclosure

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

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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 2	0 5 0 0 0 2 3 7	1 of 0 7

Title (4)

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

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 9	2 8	9 2	9 2	--- 0 3 1	--- 0 2	1 0	2 2	9 2	N/A	
									N/A	

OPERATING MODE (9)		THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIRMENTS OF 10CFR (Check one or more of the following) (11)							
POWER LEVEL (10)	0 9 4		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
M. Andjelic, Local Leak Rate Test Coordinator Ext. 2366	AREA CODE 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	I K	S M V	S 2 1 2	YES	X	B B	I S V	H 0 3 7	YES
X	L D	I S V	C 6 8 4	YES					

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)

At approximately 1910 hours on September 28, 1992 with Unit 2 at 94% power, the performance of Special Procedure 92-9-115 Revision 0 (Local Leak Rate Testing of Primary Containment Isolation Valves During Reactor Operation) identified the Outboard Drywell Air Sample Valve 2-8501-5B to be leaking 74.28 scfh. This leakage, when added to the total primary containment maximum pathway leakage rate, brought the total leakage for Type B and C testable penetrations to 435.64 scfh. Revision 0 of this report was submitted since the value exceeded an administrative leakage limit of 85% of 0.6L_a (415.18 scfh) which was established as a condition of being granted a schedular exemption from the interval required by 10CFR 50, Appendix J.

Inboard Drywell Air Sample valve 2-8501-5A was immediately closed and challenged with a local leak rate test, which yielded a leakage rate of 2.20 scfh. Valve 2-8501-5B was declared inoperable, and valve 2-8501-5A was taken Out-of-Service in the closed position. Additional testing performed on December 11, 1992 during a Unit 2 maintenance outage identified valves 2-4799-530, 2-2599-23B, and 2-2599-24B to be leaking an undetermined amount. As a result of this leakage, the Technical Specification limit for primary containment leakage, 0.6La, was exceeded.

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:

Failure of The Outboard Drywell Air Sample [IK] Valve 2-8501-5B During its 24-Month Local Leak Rate Testing Surveillance Due to Improper Valve Seating

A. CONDITIONS PRIOR TO EVENT:

Unit: 2 Event Date: September 28, 1992 Event Time: 1910 hours

Reactor Mode: N Mode Name: Run Power Level: 94%

Reactor Coolant System (RCS) Pressure: 994.9 psig

B. DESCRIPTION OF EVENT:

At approximately 1910 hours on September 28, 1992 with Unit 2 at 94% power, the performance of Special Procedure 92-9-115 Revision 0 (Local Leak Rate Testing of Primary Containment Isolation Valves [JM] During Reactor Operation) identified the Outboard Drywell Air Sample Valve 2-8501-5B to be leaking 74.28 scfh. This leakage, when added to the total primary containment maximum pathway leakage obtained through Type B and C local leak rate testing, did not exceed the Technical Specification maximum pathway leakage limit of $0.6L_a$ (488.45 scfh). However, an Administrative limit of 85% of 0.6 L_a (415.18 scfh) was exceeded since the sum of this new leakage rate caused the total Type B and C leakage rate to be 435.64 scfh. This administrative limit was established as a condition of being granted a one-time schedular exemption from the 24-month Unit 2 Type B and C local leak rate test interval required by 10 CFR 50, Appendix J.

Upon identification of the excessive leakage, the Station Control Room Engineer (SCRE) was notified of the event and the Inboard Drywell Air Sample Valve 2-8501-5A was requested to be closed. In order to verify that primary containment integrity could still be maintained, Outboard Drywell Air Sample Valve 2-8501-5B was opened and leakage past valve 2-8501-5A was quantified by performing an additional local leak rate test. The local leak rate test on Inboard Drywell Air Sample Valve 2-8501-5A yielded a leakage rate of 2.20 scfh. The new sum of this pathway's leakage rate, when added to the current sum of Type B and C leakage, caused the new total leakage rate to be 363.56 scfh. The SCRE was notified that with valve 2-8501-5A closed, primary containment integrity could be maintained. At 1910 hours on September 28,1992 Outboard Drywell Air Sample Valve 2-8501-5B was declared inoperable and Inboard Drywell Air Sample Valve 2-8501-5A was administratively taken Out-Of-Service in the closed position. This ensured that primary containment integrity could be maintained by preventing the inadvertent opening of the inboard valve. A courtesy ENS red phone notification was made at 2006 hours on September 28, 1992 to provide notification that the administrative limit for maximum pathway primary containment leakage, established in the schedular exemption granted by NRR, had been exceeded.

Work Request (WR) 12542 was previously submitted on September 18, 1992 to reduce the leakage past valve 2-8501-5B since the as-left leakage rate obtained during the D2R12 Refuel Outage was 4.0 scfh. This work request was upgraded to a B1 priority on September 28, 1992. A work analyst began preparing a package to repair valve 2-8501-5B that evening.

On October 2, 1992 maintenance was completed on valve 2-8501-5B. The valve actuator spring and diaphragm were replaced and the spring tension required to open the valve was reset. On October 6, 1992 an as-left local leak rate test was performed. This local leak rate test yielded a successful leakage rate of 3.24 scfh. The valve was tested for proper stroke time and work package 12542 was approved for completion. Valve 2-8501-5B was declared operable and returned to service at 1830 hours on October 6, 1992.

An additional valve failure was identified on December 11, 1992 with Unit 2 shutdown for a maintenance outage. At approximately 1805 hours, the performance of Dresden Technical Surveillance (DTS) 1600-01 revision 15 (Local Leak Rate Testing (LLRT) of Primary Containment Isolation Valves) identified the Drywell Pneumatic Supply [LD] Check Valve, 2-4799-530, to be leaking an undetermined amount. This leakage exceeded the maximum pathway leakage limit of 0.6La (488.452 scfh) established in Technical Specification 3.7.A.2.b(2)(a). The leakage rate for this volume was reduced to 5.8 scfh when the vent path downstream of 2-4799-530 was isolated. The Shift Engineer and the SCRE were notified of the event and Work Request 14718 was written to reduce the leakage past the check valve.

On December 12, 1992 Mechanical Maintenance inspected, cleaned, and performed a seat contact check on the valve. The contact check indicated complete seat contact; however, a portion of the seating surface did not indicate strong contact. The valve was assembled and an informational LLRT was performed. This test yielded an unsatisfactory leakage rate of 84.6 scfh. The check valve was replaced and a final post-maintenance leakage rate of 13.0 scfh was obtained.

Subsequent local leak rate testing during the maintenance outage identified the Atmosphere Containment Atmosphere Dilution (ACAD) system [BB] Outboard Check Valves 2-2599-23B and 2-2599-24B to be leaking an undetermined amount. When the vent paths down-stream of the check valves were isolated, the leakage rates were reduced to 0.0 scfh for the 2-2599-23B volume and to 0.18 scfh for the 2-2599-24B volume. The Shift Engineer and the SCRE were notified of the events and work requests 14748 and 14750 were written to inspect and repair check valves 2-2599-23B and 2-2599-24B.

Both ACAD check valves were disassembled and inspected on December 12, 1992. The valves were cleaned, cycled, and reassembled. Post-maintenance leak rate tests yielded leakage rates of 2.2 scfh for the 2-2599-23B volume and 0.82 scfh for the 2-2599-24B volume.

One additional failure occurred during the length of the 10 CFR 50, Appendix J exemption period; however, the leakage past this valve did not cause the station to exceed the administrative limit for maximum pathway leakage (85% of 0.60 L_a. On November 11, 1992 the 2A Containment Atmosphere Monitor Analyzer Sample Outlet Check Valve 2-2499-28A was found to be leaking 58.46 scfh. This leakage, when added to the current maximum pathway leakage rate, brought the total Type B and C leakage to 400.20 scfh. Since the leakage past this valve exceeded Station limits, the valve was isolated from primary containment through the Station's Out-Of-Service program and inspected on November 19, 1992 under (WR) 14085. The check valve inspection identified a build-up of debris under the valve seat which prevented full closure. The

valve internals were cleaned and the valve was reassembled. A final LLRT was performed on November 20, 1992 and yielded a leakage rate of 3.60 scfh.

C. APPARENT CAUSE OF EVENT:

This report is being submitted since both the Technical Specification limit of 0.6La (488.452 scfh) and the administrative limit of 85% of 0.6 La was exceeded during the operating cycle. The administrative limit was established as a condition for being granted a one-time schedular exemption from the 24-month Type B & C local leak rate test interval required by 10 CFR 50, Appendix J.

The apparent cause of the unsatisfactory leakage past Outboard Drywell Air Sample Valve 2-8501-5B has been attributed to inappropriate valve closing force due to inadequate actuator spring force. The actuator spring was replaced and both the original spring and replacement spring were tested. A comparison of the spring constants indicated the original spring constant to be slightly lower than the replacement spring. Local leak rate testing records dating back to 1983 indicate no previous failures of valve 2-8501-5B. In addition, maintenance records indicate that no repairs have been performed on the valve or operator.

An inspection of the Drywell Pneumatic Supply Check Valve, 2-4799-530, revealed that the seating surfaces were dirty and had become oxidized. The Mechanical Maintenance Department cleaned the seats of the valve and checked the seat contact with a bluing compound. The seat contact check revealed that a section of the valve seating surface did not indicate strong contact. The check valve inspection and contact check determined the leakage to be caused by debris preventing the check valve to close and oxidation causing the seating surfaces to wear and cause a pathway for leakage. The root cause for the source of the debris and oxidation can be attributed to system conditions (debris from within the piping and system humidity).

The ACAD Check Valves 2-2599-23B and 2-2599-24B were inspected to determine the cause of the leakage. Both check valves are a lift-type and were found with their pistons corroded to the guide. The seat on the piston of the 2-2599-23B valve was found to be satisfactory; however, the seat in the body was corroded and covered with debris. Both the piston and the valve body seats were found to be corroded and covered with debris on the 2-2599-24B valve. The cause of each valve failure was due to the corroded piston not allowing the valve to close properly. The cause of the corrosion, which prevented the piston from moving freely, is from moist air, originating from the torus, drywell, and ACAD compressor, condensing in the lines. Neither the check valves nor the attached piping is heat traced.

An inspection of the 2A Containment Atmosphere Monitor Analyzer Sample Outlet Check Valve 2-2499-28A identified debris under the valve seat. This collection of debris can be attributed to debris from within the system and moisture condensing in the piping downstream of the Containment Atmosphere Monitor.

D. SAFETY ANALYSIS OF EVENT:

The safety significance of the Outboard Drywell Air sample Valve 2-8501-5B failure is considered minimal since the Inboard Drywell Air Sample valve 2-8501-5A had a measured leakage rate of 2.20 scfh. Therefore, when adding the minimum pathway leakage rate of this pathway (2.20 scfh) to the current sum of Type B and C leakage, a combined leakage rate of 363.56 scfh would be obtained. This leakage rate is well below the leakage rate limits used in

calculating the maximum off-site dose rates established in 10 CFR 100. If the maximum pathway leakage rate for this pathway (74.28 scfh) were added to the current sum of Type B and C leakage, a combined leakage rate of 435.64 scfh would be obtained. This leakage rate would also be well below the leakage rate limits used in calculating the maximum off-site dose rates established in 10 CFR 100.

Likewise, the safety significance of the leakage past check valves 2-4799-530, 2-2599-23B, and 2-2599-24B is also considered minimal since their redundant isolation valves 2-4722, 2-2599-2B, and 2-2599-3B had leakage rates of 5.8 scfh, 0.0 scfh, and 0.18 scfh, respectively. Therefore, the actual through-leakage out of the primary containment pathways would also be below the amount used in calculating the maximum off-site dose rates established in 10 CFR 100.

E. CORRECTIVE ACTIONS:

The corrective action for the repair of valve 2-8501-5B was to verify that the spring actuator provided the required closing force needed to isolate flow. This was accomplished by replacing and adjusting the actuator spring to the vendor specification.

This valve was again tested during the D2R13 refuel outage and a leakage rate of 2.43 scfh was obtained. Long term air operated valve and actuator maintenance and diagnostic programs are in the process of being developed by corporate engineering.

Drywell Pneumatic Supply Check Valve 2-4799-530 was inspected, cleaned, and had a seat contact check performed. An informational local leak rate test was performed and yielded a leakage rate of 84.6 scfh. The check valve was then replaced and a final post-maintenance leakage rate of 13.0 scfh was obtained. Work Request 18155 was initiated to replace the valve with a safety-related valve specifically designed to seal against low pressure air.

The Unit 2 Drywell Pneumatic Supply Check valve 2-4799-530 was replaced during the D2R13 refuel outage with an in-line check valve which contains a Viton seat. A final LLRT was performed and a leakage rate of 1.75 scfh was obtained. Work Request 18923 was written to replace the Unit 3 valve during the D3R13 refuel outage (237-180-92-26001-BS1). Previous local leak rate test data back to 1983 indicate no failures for either Unit 2 or Unit 3.

ACAD Check Valves 2-2599-23B and 2-2599-24B were inspected and cleaned. Post-maintenance leak rate tests were performed and yielded leakage rates of 2.2 scfh for the 2-2599-23B volume and 0.82 scfh for the 2-2599-24B volume. Local leak rate testing records dating back to 1983 indicate five failures between the 8 check valves in service in the Unit 2 and Unit 3 ACAD Systems. In order to reduce the number of failures for this valve application, the replacement of this carbon steel valve with one which is less susceptible to corrosion (stainless steel) was evaluated. Replacement check valves were identified by Dresden's Site Engineering Department; however, the Station Technical review board determined that the current check valves should be maintained since the ACAD system will most likely be replaced with a NCAD system in the future.

Corrective actions for eliminating debris and corrosion in the Containment Atmosphere Monitoring system (CAM) are already in place. Previously the CAM system ran continuously; however, surveillance procedures have been changed within the last operating cycle which allow the system to operate for a short

period of time each month. This reduction in operating time will reduce the amount of debris and moisture introduced into the system.

F. PREVIOUS OCCURRENCES:

The most recent occurrences in which the primary containment Type B and C Leakage totals exceeded the Technical Specification Limits, and the ACAD check valves contributed significantly to the leakage, are documented in the following Licensee Event Reports.

LER/Docket Numbers	Title
88-018-1	Leak Rate Limits Exceeded. Valve 2-8501-5A was disassembled and inspected under Work Request 80052. Upon disassembly, foreign material was found on the seating surface. The valve seat was lapped and the valve internals cleaned. Upon valve reassembly, a final LLRT was performed yielding a leakage rate of 0.20 scfh.
88-018-1	Leak Rate Limits Exceeded. Check Valve 2-2599-23B was disassembled and inspected per Work Request 79715. Upon disassembly, the valve plug was found stuck in the open position. The valve internals were cleaned and proper operation of the valve plug was verified prior to reassembly.
91-007	Type B & C Containment Local Leak Rate Testing Limit Exceeded. Check Valve 3-2599-24A was disassembled and inspected under Work Request 03765. Corrosion was identified on the valve guide and seats. The valve internals were cleaned and the seating surfaces polished. The valve stroked freely after cleaning.
88-004	Type B and C Local Leak Rate Test Limit Exceeded Due to Leakage Through Primary Containment Isolation Valves. The cause of the excessive leakage from the ACAD Check Valve 3-2599-23B has been attributed to a small amount of foreign material on the valve seating surfaces coupled with slightly worn valve seats. The valve was cleaned and the seating surfaces were lapped.

G. COMPONENT FAILURE DATA:

An industry wide-Nuclear Plant Reliability Data System (NPRDS) data base search revealed seven reported failures for the Crane Model 76E check valve. Six of the failures were caused by abnormal wear, and one failure was caused by improper maintenance. All valves with failures reported were used as discharge check valves for diesel generator starting air compressors and not as containment isolation valves.

An industry-wide NPRDS data base search revealed twenty-one reported failures for the Hancock Model 5580 check valve. Four failures were attributed to corrosion, causing the plug to seize in the open position. None of the failures reported were used in similar system applications.

An industry wide NPRDS search revealed no component failure data for valve the actuator used on valve 2-8501-5B.