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 9

Title (4)

TYPE B AND C PRIMARY CONTAINMENT LOCAL LEAK RATE TESTING LIMIT EXCEEDED DUE TO LEAKAGE PAST HEAD COOLING INLET ISOLATION VALVE 2-205-2-4.

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 1	2 1	9 3	9 3	—	0 0 2	—	0 1	0 2	2 2	9 3	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	
Michael Andjelic, LLRT 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	B O	1 S V Y	L 2 0 0	Y					

SUPPLEMENTAL REPORT EXPECTED (14)			Expected Submission Date (15)	Month	Day	Year
X	YES (If yes, complete EXPECTED SUBMISSION DATE)	NO		1 2	1 5	9 3

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

On January 21, 1993 with Unit 2 in a refuel outage, the performance of Dresden Technical Surveillance (DTS) 1600-01, Local Leak Rate Testing Of Primary Containment Isolation Valves, identified the Head Cooling Inlet Isolation Valve 2-205-2-4 to be leaking an undetermined amount. This 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). Once the leakage rate was recorded, the valve was again verified to be in the fully closed position. The measured leakage rate dropped to 3.0 scfh upon increasing the seating force. The valve operator repaired under Work Request 10353, which reduced the leakage to 2.84 scfh.

The safety significance of the leakage past valve 2-205-2-4 has been considered to be minimal since the redundant Head Cooling Isolation Valve 2-205-27 leaked 3.31 scfh; therefore, the total through leakage out of the penetration, on a minimum pathway basis, was 3.31 scfh. The total as-found minimum pathway leakage (Type A test) was 2.3718 wt%/day which exceeded Technical Specification (3.7.A.2) limit of 1.2 wt%/day. Calculations have been performed to prove this leakage did not exceed 10 CFR Part 100 limits.

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 Exceeded Due To Leakage Past Head Cooling Inlet Isolation Valve 2-205-2-4

A. CONDITIONS PRIOR TO EVENT:

Unit: 2 Event Date: January 21, 1993 Event Time: 0000 hrs.

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

Reactor Coolant System Pressure: 0 psig

B. DESCRIPTION OF EVENT:

On January 21, 1993 with Unit 2 in a refuel outage, the performance of Dresden Technical Surveillance (DTS) 1600-01, Local Leak Rate Testing Of Primary Containment Isolation Valves, identified the Head Cooling Inlet Isolation Valve 2-205-2-4 to be leaking an undetermined amount. This 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). Once the leakage rate was recorded, the valve was again verified to be in the fully closed position. The actuator was manually engaged and closed with increased force. The measured leakage rate dropped to 3.0 scfh upon increasing the seating force.

The Shift Engineer was notified that the leakage past the Head Cooling Inlet Isolation Valve 2-205-2-4 had caused the total as found 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. The valve operator was repaired under Work Request 10353 which reduced the leakage to 2.84 scfh.

Additional as-found Local Leak Rate Testing of the remaining primary containment pathways identified nineteen volumes which required repairs or adjustments. The sum of all as-found Type B and C leakage calculated on a minimum pathway basis and the back correction penalty which accounts for repairs made to Type B and C volumes during the outage, not including back correction for non-vented pathways, were added. This value when added to the 95% upper confidence leak rate, measured during the Type A Integrated Leak Rate Test, along with compensation for sump level changes and non-vented systems, caused the as-found Type A leakage rate to be 2.3718 wt%/day. This value exceeded the leakage limit of 1.2 wt%/day ($0.75L_a$) stated in Technical Specification 3.7.A.2.

95% UCL Leak Rate (measured during the ILRT)	0.8184 wt%/day
Back Correction Penalty (due to repairs)	0.3225 wt%/day
Penalty For Non-Vented Systems	1.2238 wt%/day
Compensation For Sump Level Changes	0.0071 wt%/day
Total	2.3718 wt%/day

A list of the test volumes which required repairs or adjustments along with their as-found maximum Pathway and minimum pathway leakage rates are given below:

VOLUME (scfh)	SYSTEM	"As-Found" Type C (Maximum Pathway) LEAKAGE RATE (scfh)	"As-Found" (Minimum Pathway) LEAKAGE RATE
2-220-62B	Feedwater	271.3 scfh	271.3 scfh
2-220-58B	Feedwater	Undetermined	271.3 scfh
2-2301-34&71	HPCI	32.54 scfh	32.54 scfh
2-1001-1A,1B, 2A,2B,&2C	SDC	458.50 scfh	229.25 scfh
2-2001-5&6	DWEDS	18.1 scfh	9.05 scfh
2-2001-105&106	DWFDS	28.7 scfh	14.35 scfh
2-1601-20B&31B	Torus Vent	Undetermined	0.20 scfh
2-1501-27B&28B	LPCI Spray	16.24 scfh	30.12 scfh
2-4722&4799-530	DW Pneumatic	11.32 scfh	5.66 scfh
2-3703&3706	RBCCW TO DW	26.08 scfh	7.86 scfh
2-1501-22A,26A & 2-1001-5A	LPCI Inj	25.7 scfh	2.02 scfh
2-4799-514	Tip Purge	40.52 scfh	40.52 scfh
2-2599-2B&23B	ACAD Purge	Undetermined	3.53 scfh
2-8501-1A&1B	DW O2 SMPL	47.55 scfh	0.10 scfh
2-1601-21,22, 55,&56	DW Vent	651.80 scfh	0.35 scfh
2-1501-25B&26B	LPCI INJ	63.81 scfh	5.06 scfh
2-1201-1,1A,2&3	RWCU	67.29 scfh	33.65 scfh

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 2-205-2-4 valve has been attributed to insufficient closing force. The valve operator was inspected and repaired under Work Request 10353. Although the cause of failure could not be identified through motor-operated valve (MOV) testing, it is suspected that the insufficient closing force is due to a weakened spring pack that controls the seating torque of the valve. A new spring pack and a grease relief kit were installed. In addition the declutch shaft, fork assembly, and declutch lever were also replaced. Diagnostic valve testing was then performed to set the actuators opening and closing force through torque switch adjustments. A final LLRT was performed and yielded a leakage rate of 2.84 scfh. LLRT records dating back to 1983 indicate no previous failures of this valve.

A summary describing the cause and corrective actions for the remaining volumes which leaked in excess of Station guidelines are contained in Section E of this report.

D. SAFETY ANALYSIS OF EVENT:

The safety significance of the leakage past valve 2-205-2-4 has been considered to be minimal since the redundant Head Cooling Isolation Valve 2-205-27 leaked 3.31 scfh; therefore, the total through leakage out of the penetration, on a minimum pathway basis, was 3.31 scfh. The safety significance of exceeding the 1.2 wt%/day limit established in Technical Specification 3.7.A.2 is mitigated by the integrity of the Secondary Containment and the function of the Standby Gas Treatment System (SGTS). The SGTS is used to maintain a slight negative pressure in the Reactor Building during accident conditions. Filters are provided in the system to remove radioactive particulates and charcoal adsorbers are provided to remove radioactive halogens which may be present in concentrations significant to environmental dose criteria.

The calculated as-found leakage rate of 2.3718 wt%/day exceeds the Technical Specification limit of 1.2 wt%/day by a factor of approximately two. Calculations which were performed for and reported in LER 90-018-1, Leakage Path Discovered During Primary Containment ILRT due to Management Deficiency, dated August 6, 1991, indicate that a leakage rate of approximately 31 wt%/day would not exceed the off-site and control room dose rates specified by the limits in 10 CFR Part 100 and General Design Criteria 19 with SGTS operable. The D2R13 as-found leakage rate of 2.3718 wt%/day is approximately 7.7% of this value. Therefore, the safety significance of this leakage is considered minimal.

E. CORRECTIVE ACTIONS:

Immediate corrective action for the failure of valve 2-205-2-4 was to repair the actuator and verify through diagnostic tetsing that the actuator closing force was within the specified range. Additionally, an as-left LLRT was performed to ensure that the leakge past the valve seat was within acceptable limits. Long term corrective actions are already in place in that Dresden Station's motor operated valve diagnostic tetsing program will evaluate the performance of motor operated valves. This program was not firmly implemented during previous Unit 2 refuel outages; however, it is now a preferred method of monitoring motor operated valve performance.

As a result of all repairs, adjustments, and modifications made to primary containment during the D2R13 refuel outage, the total as-left maximum pathway leakage, as measured through Type B and C Local Leak Rate Testing, was 307.59 scfh. This value is 63% of the Technical Specification limit of 488.452 scfh. The as-left minimum pathway leakage rate, as measured through the Integrated Leak Rate Test, was 0.9694 wt%/day and is less than the 1.2 wt%/day limit specified in the Technical Specification.

A summary of the repairs, adjustments, and final leak rate testing results for volumes which exceeded Station guidelines for leakage along with any modifications made to containment pathways are listed below:

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

2-220-62B

Outboard B Feedwater Line Check Valve 2-220-62B was disassembled and inspected under Work Request 99084. An inspection of the valve internals revealed rust residue on the O-ring which is designed to seal between the disc/seat assembly and the valve body, and on the machined surface of the valve body which mates with the seal. This rust residue is indicative of leakage past the O-ring seal. The seat/disc assembly was removed and bench tested to verify the integrity of the seating surfaces. A LLRT was performed and yielded a leakage rate of 0.50 scfh. The disc/seat assembly was reinstalled into the valve body along with a new O-ring seal. A final LLRT was performed and an as-left leakage of 2.14 scfh was obtained. LLRT records dating back to 1983 indicate one other failure of this valve. Future corrective actions to prevent leakage past the disc/seat assembly and the valve body have been approved by the Station Modification Review Committee and will be in place for the next refuel outage. The corrective actions include machining the valve body to accept a metalic gasket, which replaces the O-ring seal, and provide additional hold down hardware for the disc/seat assembly. (237-180-93-00202)

2-220-58B

Inboard B Feedwater Line Check Valve 2-220-58B was disassembled and inspected under Work Request 10038. An inspection of the valve internals revealed approximately 0.003" clearance between the valve's hinge pins and the bushings. The disc/seat assembly to valve body mating surface showed indications of leakage past the the O-ring seal in the area between 120 and 150 degrees. Additionally, some scratches were identified in the pressure seal ring area on the valve body. The valve's disc/seat assembly was replaced since the hinge pin to bushing clearances were out of specification. A new O-ring was also installed upon reassembly. The scratches in the valve's pressure seal ring area were bored out and the surface built up and machined to the correct specifications. A final LLRT was performed and an as-left LLRT yielded a leakage rate of 3.52 scfh. LLRT records dating back to 1983 indicate two failures. Future corrective actions to prevent leakage past the disc/seat assembly and the valve body have been approved by the Station Modification Review Committee and will be in place for the next refuel outage. The corrective actions include machining the valve body to accept a metalic gasket, which replaces the O-ring seal, and provide additional hold down hardware for the disc/seat assembly. (237-180-93-00202)

2-2301-34 &71

Unit 2 HPCI Drain Pot Drain To Torus Stop Check Valve, 2-2301-71, Unit 2 HPCI Drain Pot Drain To Torus Check Valve 2-2301-34 were disassembled under work requests 15415 and 14043 respectively. Valve 2-2301-71 is not an Appendix J isolation valve; however, it is used as a boundary isolation valve during the LLRT for the 2-2301-34 valve. Troubleshooting of the 2-2301-34 valve identified the majority of the 32.54 scfh of leakage past the 2301-71 valve. This valve was removed from the line and replaced with a like for like valve since a blue check of the valve's seating surfaces revealed poor contact on one side. This damage is thought to be caused by excessive torquing of the handwheel.

Valve 2-2301-34 was disassembled and an inspection of the valve's internals was performed. A "blue check" of the valve seats revealed good contact. The valve internals were cleaned and the valve was reassembled. An as-left LLRT was performed and yielded a leakage rate of 14.17 scfh. LLRT records dating back to 1983 indicate one previous failure for each valve.

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

2-1001-1A &1B — Shutdown Cooling Inboard Isolation Valve 2-1001-1A was disassembled, inspected, and repaired under work request 10051. Upon disassembly, an inspection of the valve's internals revealed poor contact between the valve seat and disc. The valve seats were lapped and a new disc was machined for a proper fit since the original disc was too small after the required machining. Shutdown Cooling Inboard Isolation Valve 2-1001-1B was disassembled and inspected under work request 10050. An inspection of the valve internals also revealed poor contact in the valve's seating area. The valve seats were lapped and the disc machined to obtain proper contact. A final LLRT was performed and yielded a leakage rate of 27.40 scfh. LLRT records dating back to 1983 indicate one falure for valve 2-1001-1A and two failures for valve 2-1001-1B. Future corrective action for these valves is to replace the valves during the next refuel outage with an Anchor Darling parallel disc style valve. This action has already been approved by the Station Modification Review Committee. (237-180-93-00203)

2-2001-5&6 — Drywell Equipment Drain Sump Valves 2-2001-5 and 2-2001-6 were replaced with a diaphragm type valve under a Station modification since chronic problems with the old inverted solid wedge style valves were attributed to grit, pumped from the sumps, scoring the valve stem and seating surfaces. The new diaphragm valves are expected to be less susceptible to problems caused from grit entrained in water being pumped from the sumps. A final LLRT yielded a leakage rate of 0.10 scfh. LLRT records dating back to 1983 indicate one previous failure for each valve; however, various packing and timing problems have been associated with these valves.

2-2001-105 &106 — Drywell Floor Drain Sump Valves 2-2001-105 and 2-2001-106 were replaced with a diaphragm type valve under a Station modification since chronic problems with the old inverted solid wedge style valves were attributed to grit, pumped from the sumps, scoring the valve stem and seating surfaces were also prevalent. The new diaphragm valves are expected to be less susceptible to problems caused from grit entrained in water being pumped from the sumps. A final LLRT was performed and yielded a leakage rate of 0.10 scfh. LLRT records dating back to 1983 indicates one previous failure for valve 2-2001-106; likewise, various packing and timing problems have been associated with these valves.

2-1601-31B — Torus To Reactor Building Vacuum Breaker 2-1601-31B was inspected and repaired under Work request 07787. The inspection revealed the seating surfaces to be in good condition; however, it was noted during the cycling of the vacuum breaker that some times the valve would not return to the fully closed position. The problem was attributed to the valve's counterweight to be out of adjustment. The valve's counterweight was adjusted to obtain repeated closings and a torque test was performed to ensure that the vacuum breaker cycled within the correct range. A final LLRT was performed and yielded a leakage rate of 0.10 scfh. LLRT records dating back to 1983 indicate no previous valve failures. Further investigation of this event is required and the cause and corrective action will be submitted in a supplement to this report.

2-1501-27B — LPCI Drywell Spray Outboard Isolation Valve 2-1501-27B was disassembled and inspected under work request 10057. The valves disc to seat contact was found to be unacceptable after a "blue check" was performed on the seating surfaces. The valve seats were lapped and the disc was stripped of its seating material and overlayed with Stellite. Additional machining of the valve disc was performed to obtain an acceptable seat to disc contact pattern. An as-left LLRT was performed and yielded a leakage rate of 1.01 scfh. LLRT records dating back to 1983 indicate no previous failures for this valve.

2-1501-28B LPCI Drywell Spray Header Inboard Isolation Valve 2-1501-28B was disassembled and inspected under Work Request 10056. Like valve 2-1501-27B, the valve's disc to seat contact was unacceptable when a blue check was performed. The valve's seats were removed and replaced with a new set. The valve disc was stripped of its surface material and overlayed with Stellite. The disc was then machined to provide for proper contact with the seats. A final LLRT was performed and yielded a leakage rate of 1.01 scfh. LLRT records dating back to 1983 indicate no previous failures for this valve.

2-4799-530 Drywell Pneumatic Inlet Check Valve To PCV 2-4722 was replaced under Work Request 18155. The original check valve was a swing-type check valve and the replacement valve was an in-line wafer type check valve with a Viton seat. This valve design was chosen to better seal against low pressure air similar to what would be experienced during post LOCA conditions and LLRT Tests. A final LLRT was performed and yielded an as-left LLRT of 1.75 scfh. LLRT records dating back to 1983 indicate one previous failure.

2-3703 RBCCW return From the Drywell Outboard Isolation Valve 2-3703 was inspected under Work Request 10060. An inspection of the valve internals revealed marginal contact. The valve's disc and seats were lapped to clean and true the surfaces. A final "blue check" was performed and the valve was reassembled. A final leakage rate of 8.16 scfh was obtained. LLRT records dating back to 1983 indicate no previous failures.

2-1501-23A Unit 2 LPCI Loop I Coolant Injection Test Connection Stop Valve 2-1501-23A exhibited significant packing leakage, 25.7 scfh, during the as-found LLRT for valves 2-1501-26A, 2-1501-22A, and 2-1001-5A. Work Request 15937 was written to repack the valve to reduce the leakage. The valve was repacked and a LLRT was performed to verify there were no additonal problems with the other valves in the test volume. During the repacking of valve 2-1501-23A it was observed that the stem was bent in the area of the packing gland area. Work Request 16261 was written to replace the valve. The valve was replaced and a final LLRT was performed and yielded a leakage rate of 4.48 scfh for the test volume consiting of valves 2-1501-26A, 2-1501-22A, and 2-1001-5A. LLRT records dating back to 1983 indicate no previous failures for this valve.

2-4799-514 TIP Purge Check Valve 2-4799-514 was replaced with a suitable valve under Work Request 16400. LLRT records dating back to 1983 indicate no previous failures for this valve.

2-2599-23B ACAD Drywell Air Purge Inlet Check Valve 2-2599-23B was disassembled and inspected under Work Request 15771. An inspection of the valve internals revealed pitting, caused by corrosion, on the piston and its guide. In additon, black debris was found on the seat and bottom of the valve. The valve internals were cleaned and the valve piston and seat were lapped to ensure proper contact. The piston guide was also cleaned and a final "blue check" was performed. The "blue check" was acceptable and the valve was reassembled. A final LLRT was performed and yielded a leakage rate of 1.10 scfh. LLRT records dating back to 1983 indicate two previous failures. The cause for the black debris found in the valve internals has been attributed to moisture in the piping system.

2-8501-1B

Torus Air Sample Valve 2-8501-1B was evaluated and adjusted under Work Request 15717. The valve's stroke length was first checked and the closing spring force was adjusted to the appropiate setting. The cause of the failure has been attributed to improper valve stroke which can be caused through normal wear and cycling. An as-left LLRT was performed and yielded a leakage rate of 0.10 scfh was obtained. LLRT records dating back to 1983 indicate no previous failures of this valve.

2-1601-55

Drywell Vent Valve 2-1601-55 was disassembled and inspected under Work Request 13780. An inspection of the valve revealed that the soft seat ring was cracked. A new seat ring was installed in the valve and the valve's operator was rebuilt. A final LLRT was performed on theis valve and other valves in the test volume and yielded a leakage rate of 0.67 scfh. Previuos LLRT data dating back to 1983 indicates no previous failures of this valve.

2-1501-25B

The unit 2 LPCI B Loop Injection Line Check Valve 2-1501-25B was disassembled and inspected under Work Request 15952. An inspection of the valve internals revealed debris located in the disc/seat contact area. In addition, a "blue check" of the valve's seating area revealed incomplete contact along a portion of the seating surface. The valve seat and disc were lapped until an acceptable "blue check" was obtained. The valve was reassembled and a final LLRT was performed. this LLRT yielded a leakage rate of 4.32 scfh. LLRT data dating back to 1983 indicates no previous valve failures.

2-1299-004

Reactor water cleanup test connection and drain valves 2-1299-4 and 2-1299-5 were removed from the piping under Work Request 16368 and bench tested after troubleshooting of the test volume indicated potential leakage through these valves. The valves were removed and bench tested to confirm and quantify the leakage past these valves. A LLRT was perfomed on the valves and indicated a leakage rate of approximatly 60 scfh. New valves were installed and a final LLRT yielded a leakage rate of 5.21 scfh.

Electrical Penetration X-203A

Electrical penetration X-203A was repaired under Work Request 15716. Leakage out of this penetration was reduced from 18.41 scfh to 10.39 scfh after being repaired with Loctite 286 sealant. LLRT records dating back to 1983 indicate no previous failures for this penetration.

Electrical Penetration X-202W

Electrical penetration X-202W was repaired under Work Request 12601 Leakage out of this penetration was reduced from 55.89 Penetration scfh to 1.65 scfh after being repaired with Loctite 286 sealant. LLRT records dating back to 1983 indicate two previous failures for this penetration.

Penetrations X-113,X-125 X-149,X-149B

In addition to the above stated repairs, bellows penetrations X-113, X-125, X-149A, and X-149B were replaced with a new design which provides an increased space between the plies. This allows the total surface of the bellows to be challenged during Type B Local Leak Rate Testing.

Penetration X-144

The CRD return line from the reactor, 2-0308-4"-A was cut and capped during the D2R13 refuel outage. This planned modification eliminates valves 2-301-95 and 2-301-98 as Appendix J primary containment isolation valves. In addition, bellows penetration X-144 was sealed from within the drywell since the CRD return line was cut and capped. This eliminates Bellows X-144 as an Appendix J Type B testable volume.

F. PREVIOUS OCCURRENCES:

LER/Docket Numbers	Title
90-009/0500237	Type B and C Primary Containment Local Leek Rate Test Requirements Exceeded Due to Leaking Isolation Valves.

G. COMPONENT FAILURE DATA:

Manufacturer	Nomenclature	Model Number	Mfg. Part Number

Component failure data will be submitted in A supplement to this report.