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 Nuclear Power Station, Unit 2	05000237	1 OF 10

TITLE (4)
Technical Specification Type B and C Leakage Limit Exceeded Due to Excessive Leakage Past High Pressure Coolant Injection Check 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
06	10	95	95	-- 018 --	01	05	14	96	None	
									FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	N
POWER LEVEL (10)	000

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

	20.2201(b)		20.2203(a)(3)(i)		50.73(a)(2)(iii)		73.71(b)
	20.2203(a)(1)		20.2203(a)(3)(ii)		50.73(a)(2)(iv)		73.71(c)
	20.2203(a)(2)(i)		20.2203(a)(4)		50.73(a)(2)(v)		OTHER
	20.2203(a)(2)(ii)		50.36(c)(1)		50.73(a)(2)(vii)		(Specify in Abstract below and in Text, NRC Form 366A)
	20.2203(a)(2)(iii)		50.36(c)(2)		50.73(a)(2)(viii)(A)		
	20.2203(a)(2)(iv)	X	50.73(a)(2)(i)		50.73(a)(2)(viii)(B)		
	20.2203(a)(2)(v)	X	50.73(a)(2)(ii)		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	BJ	ISV	C283	Yes		X	SJ	ISV	C665	Yes
X	AD	ISV	C635	Yes		X	BO	ISV	C665	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 15 single-spaced typewritten lines) (16)

At approximately 0830, on June 10, 1995, with Unit 2 shutdown for Refuel Outage D2R14, the performance of a Dresden Technical Surveillance identified the High Pressure Coolant Injection (HPCI) System Turbine Exhaust to Suppression Pool Check Valve 2-2301-45 to be leaking more than the test equipment could measure. When the valve's leakage was added to the existing maximum pathway leakage rate, the maximum pathway leakage rate Technical Specification limit for Type B and C primary containment leakage was exceeded. The safety significance of the leakage past the 2-2301-45 was considered to be minimal since the additional leakage out of containment, on a minimum pathway basis, was 0 scfh from the inboard isolation Stop Check Valve 2-2301-74 and would not cause the maximum off-site dose rates established in 10 CFR 100 to be exceeded. The check valve was removed, inspected, replaced and Local Leak Rate Tested prior to unit startup. This supplement contains the root cause, corrective actions taken and results of an NPRDS search for all test volumes that exceeded administrative local leakage rate limits during Refuel Outage D2R14.

NRC FORM 366B (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)
FAILURE CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Dresden Nuclear Power Station, Unit 2	05000237	95	-- 018 --	01	2 OF 10

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	BR	ISV	C665	Yes					
X	CE	ISV	C665	Yes					
X	BM	ISV	C665	Yes					
X	BB	ISV	W255	Yes					
X	CC	ISV	C665	Yes					

EVENT IDENTIFICATION:

Type B and C leakage limit exceeded due to excessive leakage past HPCI check valve,

A. PLANT CONDITIONS PRIOR TO EVENT:

Unit: 2 Event Date: 06/10/95 Event Time: 0830 hrs

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

Reactor Coolant System Pressure: 0 psig

B. DESCRIPTION OF EVENT:

This LER is being submitted pursuant to 10 CFR 50.73(a)(2)(i) which requires the reporting of any operation or condition prohibited by the plant's Technical Specifications.

This LER is also submitted pursuant to 10 CFR 50.73(a)(2)(ii) which requires reporting any event or condition that resulted in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded.

At approximately 0830, on June 10, 1995, with Unit 2 shutdown for Refuel Outage D2R14, the performance of Dresden Technical Surveillance (DTS) 1600-01, Local Leak Rate Testing Of Primary Containment Isolation Valves, identified the HPCI Turbine Exhaust to Suppression Pool Check Valve 2-2301-45 to be leaking more than the test equipment could measure. When the valve's leakage was added to the existing maximum pathway leakage rate, the maximum pathway leakage rate limit 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) was exceeded.

The Unit Supervisor was notified of the event and a Performance Improvement Form (PIF) was written to report a condition prohibited by the plant's Technical Specifications.

A list of additional valves which failed the administrative leakage limit during the outage is provided below. The test volumes which required repairs or adjustments and the as-found maximum pathway and minimum pathway leakage rates are included in the table. The minimum pathway leakage rate reflects the actual leakage which would occur under design basis accident conditions.

VOLUME	SYSTEM	(Maximum Pathway) LEAKAGE RATE	(Minimum Pathway) LEAKAGE RATE
2-220-44 & 45	Recirc Sample	12.21 scfh	0.10 scfh
2-220-57B & 62B	Feedwater	Undetermined	1.02 scfh
2-1001-1A,1B,2A, 2B & 2C	Shutdown Cooling	64.33 scfh	32.17 scfh
2-1101-1 & 16	SBLC	12.15 scfh	4.24 scfh
2-1201-1,1A,2,3	RWCU	39.60 scfh	0.10 scfh
2-1402-6A & 25A	Core Spray	38.84 scfh	1.03 scfh
2-2599-2A & 23A	ACAD	31.49 scfh	0.10 scfh
2-3706	RBCCW	36.36 scfh	15.80 scfh

Elec Pen X-202BB	CRD Indication	20.83 scfh	10.42 scfh
Bellows X-108A	Iso Condenser	5.84 scfh	2.92 scfh
Bellows X-116A	LPCI	3.04 scfh	1.52 scfh
Bellows X-116B	LPCI	2.22 scfh	1.11 scfh
Bellows X-124	RBCCW	1.62 scfh	0.81 scfh
Bellows X-126	Cnmt. Purge	2.84 scfh	1.42 scfh
Shear Lug Insp. Hatch No. 8	Containment	5.50 scfh	2.25 scfh

PIFs (Performance Improvement Forms) were written to document each of these valve leakage failures.

C. CAUSE OF EVENT:

The cause of previous LLRT failures for the 2(3)-2301-45 check valve was that the two disks of the check valve would periodically cycle until damage to the seating surface resulted. The disks would cycle (not remain full open) during Operating Department Surveillances due to the inadequate exhaust flow from the HPCI turbine. This cycling would occur mostly during low HPCI turbine speeds.

Two actions were implemented to reduce the disk cycling. First, the closing springs of the C & S Valve Co. dual disk check valve were replaced with ones having a torque rating of 35 in.-lbs. The original-supplied closing springs had a torque rating of 109 in.-lbs. The purpose of this change was to allow the disks to fully open during low speed HPCI turbine operation. By reducing the closing spring torque, the amount of turbine exhaust flow required to maintain the disks in a full open position was reduced. Secondly, the duration of low speed HPCI turbine operation was reduced. This was performed through changes to Operating Department procedures. Both of these changes reduced the check valve disk cycling.

When the 2-2301-45 valve was removed from the system, the inspection revealed the seats to be in good condition. The inspection also indicated that one of the two disks would not go fully closed. As the disks were released from the full open position, one disk would stop midcycle leaving a gap as large as one inch between the seat and the disk. This gap is large enough so that flow through a 3/4" test tap would pass through the gap and not provide adequate force to fully close the disk thus resulting in a condition where LLRT test equipment could not measure this leakage rate. The manufacturer, C & S Valve Co., was contacted for assistance and concurred that the disk would not go fully closed with relatively low reverse flow (3/4" test tap versus 24" check valve) and low closing spring torque (35 in.-lbs).

The inspection also found internal corrosion on the body of the carbon steel dual disk check valve. This is due to condensed steam lying in the bottom of the check valve. One of the horizontally mounted check valve's dual disks rides on the bottom of this check valve. Lower portions of this disk and the disk lug assembly were also found corroded. This corrosion results in an added frictional force which needs to be overcome by the closing spring in order to close the check valve.

Therefore, the root cause of the failure of the 2-2301-45 valve was ineffective corrective actions implemented from a previous LLRT failure in that low torque

springs may keep the valve from cycling, however, the valve may not go closed with no flow or even low reverse flow through the valve.

A summary describing the cause and corrective actions for the remaining volumes which leaked in excess of Station administrative leakage limits during Refuel Outage D2R14 are contained in Section E of this report.

D. SAFETY ANALYSIS:

The safety significance of the leakage past the 2-2301-45 was considered to be minimal since the additional leakage out of containment, on a minimum pathway basis, was 0 scfh from the inboard isolation Stop Check Valve 2-2301-74 and would not cause the maximum off-site dose rates established in 10 CFR 100 to be exceeded.

A 10 CFR 50, Appendix J Type A ILRT was performed to challenge Unit 2 Containment Integrity on February 29, 1996 as part of Refuel Outage D2R14. The as-left minimum pathway leakage was calculated to be 0.44198 wt%/day. The calculated as-found minimum pathway leakage rate was 0.54296 wt%/day. Both of these values are well below the Technical Specification limit for primary containment minimum pathway leakage of 1.2 wt%/day. Since 10 CFR 100 limits are not exceeded with Tech Spec limit leakage, the 10 CFR 100 limits would not have been exceeded due to either as-found or as-left minimum pathway leakage. Therefore, the safety significance of these Primary Containment Isolation Valves leaking in excess of Station administrative limits is considered minimal.

E. CORRECTIVE ACTIONS:

The repairs to the HPCI Turbine Exhaust Check Valve, 2-2301-45, included replacement of the check valve, even though there was no damage to the seating surfaces, and return to the closing spring with the higher torque value. The as-left Local Leak Rate Test yielded a leakage rate of 3.69 scfh. A reduction in the duration of low speed HPCI Turbine operation will continue to minimize cycling of the check valve and the return of a tighter spring will ensure valve closure.

The HPCI Turbine Exhaust Check Valve 2-2301-45 was removed, inspected, replaced, and Local Leak Rate Tested prior to unit start up. (Ref. NTS #237-180-95-01801)

This LER supplement contains the cause of and the repairs performed for D2R14 valve LLRT failures as well as the results of the as-left LLRTs. (Ref. NTS #237-180-95-01802)

Valve component engineers were established at Dresden Station in order to increase valve performance. This was done through the use of diagnostic testing and preventive maintenance so as to minimize the amount of corrective maintenance needed to be performed. The Valve Engineers were tasked with developing and implementing plans for increasing the performance of a specific type of valve; motor-operated valves, air-operated valves, check valves and manual and relief valves. This focused attention resulted in a reduction of LLRT valve failures from 17 failures in the previous refuel outage D2R13 to 9 failures in D2R14.

A summary of the root cause, repairs, adjustments, and as-left leak rate testing results for volumes which exceeded Station administrative leakage limits are listed below:

2-220-45 Recirc System Sample Valve 2-220-45 was cut out and replaced under Work Request 940097442. An as-left LLRT was performed which yielded a leakage rate of 3.93 scfh. LLRT records dating back to 1983 indicate one previous failure of this valve, however, there was no valve repair performed. Valve replacement is considered satisfactory corrective action based on failure records.

2-220-62B Feedwater Outboard Check Valve 2-220-62B was disassembled and inspected under Work Request 950018354. Reddish debris was found on the seat, however, seat to disk contact was tight and 360 degrees complete. Valve internals showed indication of leakage between the valve body and the seat/disk assembly. The "O" ring between the seat/disk assembly and the valve body was in poor condition. Valve design was determined to be the cause of the degradation of the "O" ring which allowed leakage between the valve body and the seat/disk assembly. Because of problems with a planned modification's design, valve refurbishment was determined as corrective action for the leak rate failure. The "O" ring was replaced, valve internals cleaned and the valve reassembled. An as-left LLRT was performed which yielded a leakage rate of 5.29 scfh. LLRT records dating back to 1983 indicate 3 previous failures of this valve occurring during the last 3 refueling outages. Required periodic LLRT testing will continue to be performed on valve 2-220-62B to determine if additional repairs are necessary.

2-1001-1A
2-1001-1B Shutdown Cooling Inboard Isolation Valves 2-1001-1A and 2-1001-1B were cut out as part of the materiel condition upgrade in order to improve valve leak rate performance and lower dose rates in the drywell (Work Request 940097078). The Crane flex wedge gate valves were replaced with an Anchor Darling dual disk gate valve. In addition, system vent valves 2-1001-47B and 2-1001-48B, located within the LLRT test boundary, were found to be leaking. These vent valves were cut out and replaced. An as-left LLRT was performed which yielded a leakage rate of 20.56 scfh. LLRT records dating back to 1983 indicate 3 previous failures of this valve.

2-1101-16 Standby Liquid Control (SBLC) Outboard Isolation Check Valve 2-1101-16 was disassembled and inspected under Work Request 940097418. The inspection revealed 360 degrees of plug to seat contact, however, light brown corrosion on the in-body seat and plug were discovered. The seat and plug were cleaned and a blue check revealed good contact. The as-left LLRT yielded a leakage rate of 0.81 scfh. LLRT records dating back to 1983 indicate no previous failures. The corrective action of cleaning is considered adequate corrective action for this failure because of the low failure history.

2-1201-3 Reactor Water Cleanup (RWCU) Outboard Isolation Valve 2-1201-3 was disassembled and inspected under Work Request 940094538. The inspection revealed that there was not a complete 360 degrees of seat to wedge contact but rather low spots on the in-body and wedge seating areas. Repairs included weld build-up and subsequent machining of the valve wedge. An as-left LLRT was performed which yielded a leakage rate of 6.10 scfh. LLRT records dating back to 1983 indicate no previous failures of this valve. Corrective actions of cleaning and refurbishing are considered adequate corrective actions for this failure because of the low failure history.

2-1402-25A Core Spray Injection "A" Inboard Gate Valve 2-1402-25A was disassembled and inspected under Work Request 940094562. The inspection revealed scratches across the top of the valve wedge and the valve seat. Repairs included a skim cut of the valve wedge to remove the scratches and dextering of the in-body valve seat. An as-left LLRT was performed which yielded a leakage rate of 2.02 scfh. LLRT records dating back to 1983 indicate no previous failures of this valve. Corrective actions of cleaning and refurbishing are considered adequate corrective actions for this failure because of the low failure history.

2-2599-2A ACAD Dilution To Drywell Inboard Valve 2-2599-2A was disassembled and inspected under Work Request 950027947. The valve internals were replaced and the valve was subjected to diagnostic testing of the air operator An as-left LLRT was performed which yielded a leakage rate of 2.66 scfh. LLRT records dating back to 1983 indicate no previous failures of this valve. Corrective action of refurbishment is considered adequate corrective action for this failure because of the low failure history.

2-3706 Reactor Building Closed Cooling Water (RBCCW) Inboard Drywell Outlet Valve 2-3706 was disassembled and inspected under Work Request 940096672. A scratch was found across the wedge seating area. This scratch was polished away, the valve stem replaced and the valve reassembled. An as-left LLRT was performed which yielded a leakage rate of 14.25 scfh. LLRT records dating back to 1985 indicate no previous failures of this valve. Corrective action of refurbishment is considered adequate corrective action for this failure because of the low failure history.

Elec Pen X-202BB Electrical Penetration X-202BB was leaking in excess of the administrative leakage limit of 10 scfh with a leakage of 20.83 scfh. This electrical penetration contains cabling with Amphenol connections for CRD indication. During the previous Refuel Outage D2R13, leakage was noted coming from between two connectors. This means that cabling insulation has cracks in it allowing for nitrogen to travel between the conductor and the insulation to the end of the cable and then escaping between connectors. Loctite injection between the connectors was not deemed an option since the vendor feared that the connector pins may get coated and not allow the CRD indication to get back to the Control Room. Since the leakage rate

has remained essentially the same during the last 6 refueling outages, no repairs are planned at this time.

Bellows X-108A — Isolation Condenser Steam Supply Piping Expansion Bellows X-108A was cut out under Work Request 930049716. As part of the bellows replacement upgrade initiative, this bellows was replaced by a single ply expansion bellow and was welded in place. A Bellows Test Enclosure was then installed. This Bellows Test Enclosure is a removable outer cover which is a non-safety protection boundary. The as-left LLRT yielded a leakage rate of 4.66 scfh.

Bellows X-116A — Low Pressure Coolant Injection (LPCI) "A" Piping Expansion Bellows X-116A was cut out under Work Request 930049715. As part of the bellows replacement upgrade initiative, this bellows was replaced by a single ply expansion bellow and was welded in place. A Bellows Test Enclosure was then installed. This Bellows Test Enclosure is a removable outer cover which is a non-safety protection boundary. The as-left LLRT yielded a leakage rate of 1.32 scfh.

Bellows X-116B — Low Pressure Coolant Injection (LPCI) "B" Piping Expansion Bellows X-116B were leaking in excess of their 0.50 scfh administrative leakage limit. Excessive leakage requires the volume between the bellows to be pressurized with helium. The bellows are then checked for integrity using helium detection equipment. These bellows were determined to have their outer ply intact and no repair was necessary. When future testing determines that neither the inner or outer plies are completely intact, the bellows will be repaired.

Bellows X-124 — Reactor Building Closed Cooling Water (RBCCW) Piping Expansion Bellows X-124 were leaking in excess of their 0.50 scfh administrative leakage limit. Excessive leakage requires the volume between the bellows to be pressurized with helium. The bellows are then checked for integrity using helium detection equipment. These bellows were determined to have their outer ply intact and no repair was necessary. When future testing determines that neither the inner or outer plies are completely intact, the bellows will be repaired.

Bellows X-126 — Containment Purge Piping Expansion Bellows X-126 was cut out under Work Request 930049714. This bellows was replaced with a design which provides for increased space between the plies, thereby allowing the total surface of the bellows to be challenged during its Type B Local Leak Rate Test. The as-left LLRT yielded a leakage rate of 0.10 scfh.

Shear Lug Insp Hatch No. 8 — The double gasket seal for the number 8 Shear Lug Inspection Hatch Cover leaked in excess of the administrative leakage limit of 5 scfh with a leakage of 5.50 scfh. Due to the extensive amount of work required to access the inspection hatch, the steady trending up of leakage and no expected excessive leakage increase during this operating cycle, this seal will be repaired. (WR 950036512). (NTS 237180950180351)

F. PREVIOUS OCCURRENCES:

LER/Docket Numbers	Title
95-011/0500249	Type B and C Leakage Limit Exceeded Due to Excessive Leakage Past HPCI Check Valve
94-022/0500237	Type B and C Leakage Limit Exceeded Due to Worn Seating Surface of HPCI Check Valve
91-007/0500249	Type B and C Containment Local Leak Rate Testing Limit Exceeded Due to HPCI Turbine Exhaust Check Valve Leakage
89-009/0500249	Local Leak Rate Testing "As Found" limit Exceeded Due to leakage From Primary Containment Valves

G. COMPONENT FAILURE DATA:

Manufacturer	Nomenclature	Model Number	Mfg. Part Number
Copes - Vulcan	Reactor Recirc Sample Valve 2-0220-45	D-Style	N/A

An industry-wide data base search revealed 3 failures for Copes - Vulcan Model D-Style globe valves installed in Reactor Recirculation Systems. Two failures were caused by wear of valve internals.

Crane Valve Co.	"B" Feedwater Line Check Valve 2-220-62B	973	N/A

An industry-wide data base search revealed 144 failures for the Crane Model 973 tilting disc check valve. Thirty two failures were attributed to failures of the "O" ring between the valve body. Most of the failures were in high temperature, high flow feedwater systems.

Crane Valve Co.	Shutdown Cooling Inlet Valve 2-1001-1A Shutdown Cooling Inlet Valve 2-1001-1B	783-UL	N/A

An industry-wide data base search revealed 26 failures of Crane Valve Co. Model 783 gate valves due to normal wear and poor seat conditions.

Crane Valve Co.	Standby Liquid Control Injection Check Valve 2-1101-16	3888-U	N/A

An industry-wide data base search revealed 7 failures of Crane Model 3888-U lift-type check valves installed in Standby Liquid Control Systems. Six failures were due to general corrosion of seating surfaces not allowing proper seat-plug contact.

Crane Valve Co.	Reactor Water Cleanup Inlet Valve 2-1001-1A	783-UL	N/A

An industry-wide data base search revealed 26 failures of Crane Valve Co. Model 783 gate valves due to normal wear and poor seat conditions.

Crane Valve Co.	Core Spray "A" Injection Valve 2-1402-25A	783-UL	N/A

An industry-wide data base search revealed 26 failures of Crane Valve Co. Model 783 gate valves due to normal wear and poor seat conditions.

C & S Valve Co.	HPCI Turbine Exhaust to Suppression Pool Check Valve 2-2301-45	N/A	N/A

An industry-wide data base search revealed 3 failures of C & S Valve Co. dual disk check valves installed in High Pressure Coolant Injection Systems. These failures were all due to normal wear of valve internals.

WKM	ACAD Inlet To Drywell Valve 2-2599-2A	N/A	N/A

An industry-wide data base search revealed 9 failures of the WKM globe valve installed in Combustible Gas Control Systems. One failure was due to worn valve internals.

Crane Valve Co.	Reactor Building Closed Cooling Water Valve 2-3706	471/2-XR	N/A

An industry-wide data base search revealed 8 failures of Crane Model 143-1/2U gate valves.