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

July 27, 1992

CWS LTR #92-463

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Licensee Event Report #90-009-2, Docket #050237 is being submitted as required by Technical Specification 6.6, NUREG 1022 and 10 CFR 50.73(a)(2)(i). This revised report provides clarification concerning the corrective action program regarding inspection/testing of primary containment bellows penetration assemblies.

L. A. Hansen for 8/13/92

C. W. Schroeder
Station Manager
Dresden Nuclear Power Station

CWS/omf

Enclosure

cc: A. Bert Davis, Regional Administrator, Region III
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 1 2

Title (4)

Type B and C Primary Containment Local Leak Rate Test Requirements Exceeded Due to Leaking Isolation Valves

Event Date (5)			LER Number (6)				Report Date (7)			Other Facilities Involved (8)	
Month	Day	Year	Year	///	Sequential Number	Revision Number	Month	Day	Year	Facility Names	Docket Number(s)
										N/A	
0 9	2 3	9 0	9 0	---	0 0 9	--- 0 2	1 0	1 6	9 0	N/A	

OPERATING MODE (9)		THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR (Check one or more of the following) (11)			
	N	___ 20.402(b)	___ 20.405(c)	___ 50.73(a)(2)(iv)	___ 73.71(b)
POWER LEVEL (10)	0 0 0	___ 20.405(a)(1)(i)	___ 50.36(c)(1)	___ 50.73(a)(2)(v)	___ 73.71(c)
		___ 20.405(a)(1)(ii)	___ 50.36(c)(2)	___ 50.73(a)(2)(vii)	___ Other (Specify
		___ 20.405(a)(1)(iii)	X 50.73(a)(2)(i)	___ 50.73(a)(2)(viii)(A)	in Abstract
		___ 20.405(a)(1)(iv)	___ 50.73(a)(2)(ii)	___ 50.73(a)(2)(viii)(B)	below and in
		___ 20.405(a)(1)(v)	___ 50.73(a)(2)(iii)	___ 50.73(a)(2)(x)	Text)

LICENSEE CONTACT FOR THIS LER (12)

Name	TELEPHONE NUMBER
	AREA CODE
M. Andjelic, Technical Staff Engineer Ext. 2366	8 1 5 9 4 2 - 2 9 2 0

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	///	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	///
X	S B	X 1 5 V	C 6 6 5	Y							
X	B M	X 1 5 V	C 6 6 5	Y							

SUPPLEMENTAL REPORT EXPECTED (14)		Expected Submission Date (15)	Month	Day	Year
___ Yes (If yes, complete EXPECTED SUBMISSION DATE)	X NO				

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

On September 23, 1990, with Unit 2 in a refuel outage during performance of Main Steam Isolation Valve (MSIV) Local Leak Rate Testing (LLRT), the A, C, and D main steam line volumes were found to be leaking in excess of the Technical Specification limit of 11.5 SCFH. Further diagnostic testing indicated that MSIVs 2-203-1A, 1D, and 2C were the leaking valves. The cause of the leakage has been determined to be wear of the valve seating surfaces. On September 25, 1990, while performing further LLRTs, outboard primary containment drywell spray valve 2-1501-27A leaked an undetermined amount. This caused the as-found total type B and C leakage rate to be in excess of the Technical Specification limit. After flushing of the valve seat, the leak rate was reduced to a minimal level. The safety significance for both events was minimal since in each case the in line isolation valves were not observed to be leaking. Of the remaining type B and C tests which demonstrated unsatisfactory leakages, the actual through leakage was determined to be minimal. On April 2, 1991, while Unit 2 was shutdown for a short maintenance outage, six drywell bellows penetrations were inspected for cracks. Of the six, through wall cracks were found at penetration X144. However, the safety significance is considered minimal due to the satisfactory performance of these bellows during the recent Integrated Leak Rate Test (ILRT). A previous event involving the as found type B and C test results exceeding the 0.6 La limit is outlined in LER #88-004 on Docket #050249.

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 Test (LLRT) Requirements Exceeded Due To Leaking Isolation Valves.

CONDITIONS PRIOR TO EVENT:

Unit: 2 Event Date: September 23, 1990 Event Time: 1800 hrs

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

Reactor Coolant System (RCS) Pressure: 0 psig

B. DESCRIPTION OF EVENT:

On September 23, 1990, with Unit 2 in a refuel outage, while performing Dresden Technical Staff Surveillance (DTS) 250-1, Main Steam [SB] Isolation Valve (MSIV) Local Leak Rate (Dry) Test, the A, C, and D MSIVs leaked 33.0 standard cubic feet per hour (SCFH), 17.8 SCFH, and 39.0 SCFH respectively. After further diagnostic testing on 9/28/90 per DTS 250-3, MSIV Local Leak Rate (Wet) Test, it was determined that inboard MSIV 2-203-1A leaked 23.5 SCFH, inboard MSIV 2-203-1D leaked 39.0 SCFH, and outboard MSIV 2-203-2C leaked 14.2 SCFH. These leakage values exceeded Technical Specification 3.7.A.2.b.(2)(c), which states that the leakage past any one Main Steam Isolation Valve shall not exceed 11.5 SCFH at a test pressure of 25 psig.

The MSIV Local Leak Rate (Dry) Test is performed by pressurizing the volume between the inboard and outboard MSIVs and monitoring the makeup flow to the volume which is required to maintain the test pressure. This test is performed with the reactor level below the Main Steam Line nozzles; consequently, both the inboard and outboard MSIVs are vented directly to the atmosphere. Because of this, it is impossible to quantify the leakage past each individual MSIV. The MSIV Local Leak Rate (Wet) Test is performed after the reactor water is flooded up to the level of the fuel pool. With this configuration, the water head pressure on the upstream side of the inboard MSIV exceeds the test pressure. Therefore, any leakage observed during the wet test can be assumed to equal the leakage past the outboard MSIV. Consequently, the leak rate of the inboard MSIV can be calculated by subtracting the "wet" test results from the "dry" test results. Attachment 1 of this report contains a table of the leakage rates obtained from the "wet" and "dry" tests, along with the calculated leakage rates for each MSIV.

Work Requests (WRs) 95359, 95360, and 95361 were subsequently initiated to repair MSIVs 2-203-2C, 2-203-1A, and 2-203-1D respectively.

On September 25, 1990, with Unit 2 still in the refuel outage, while performing DTS 1600-1, Local Leak Rate Testing (LLRT) of Primary Containment Isolation Valves, on the Low Pressure Coolant Injection (LPCI) [BM] primary containment drywell spray valves 2-1501-27A, 28A, it was discovered that the outboard isolation valve (2-1501-27A) leaked an undetermined amount. The LLRT on these valves is performed by pressurizing the volume of piping between valves 2-1501-27A and 28A to 48 psig and then monitoring the makeup flow required to maintain that pressure. In this case, however, the testing personnel were unable to pressurize the piping volume to the required test pressure. Consequently, the as-found Type B and C leak rate for this volume was documented as undetermined. This resulted in the total as found Type B and C leakage rate exceeding the Technical Specification 3.7.A.2.b.(2)(a) limit of 488.452 SCFH. The total Type B and C leakage rate is determined utilizing the maximum pathway methodology. This methodology is based on the assumption that the better valve of a dual valve set is unable to perform its containment isolation function.

Inspections of the upstream and downstream vents verified that outboard valve 2-1501-27A was the leaking valve. Further investigation suggested the cause of leakage was due to a build up of foreign material in the valve seat area. Initial troubleshooting involved flushing the valve thoroughly with clean demineralized water. Subsequent to this, a retest was performed which yielded a satisfactory leakage rate of 0.4 SCFH. As the inboard valve was not disturbed before the retest, it was clearly intact in the as-found condition.

In addition to the MSIVs and the LPCI isolation valve 2-1501-27A, ten type C and two type B tests required repairs and adjustments due to unsatisfactory leakage rates. These test volumes and their corresponding "as found" leakage rates are listed below:

Test Volume	System	"As Found" Type B and C (Maximum Pathway) Leakage Rate (scfh)	"As Found" Type A (Minimum Pathway) Leakage Rate (scfh)
2-205-2-7	Head Spray	undetermined	1.0
2-220-57A, 58A	Feedwater [SJ]	51.2	27.9
2-220-57B, 62B	Feedwater	undetermined	0.3
2-301-160A, 161A	Scram Discharge Volume [AA]	73.0	10.3
2-1001-1A,1B,2A,2B,2C	Shutdown Cooling	232.6	30.6
2-1301-3, 4	Isolation Condenser [BL]	114.9	0.1
2-1501-18B, 19B	LPCI [BO]	78.4	7.3
2-2001-5, 6	Drywell Equipment Drain [WK]	35.0	17.5
2-2301-34, 71	High Pressure Coolant Injection (HPCI) [BJ]	34.6	5.1

2-3702, 3799-126	Reactor Building Closed Cooling Water (RBCCW) [CC]	131.1	12.5
Electrical Penetration - X202BB		22.0	11.0
Drywell Bellows - X113		50.0	0.0

On April 2, 1991, with Unit 2 shutdown for a short maintenance outage, while inspecting the bellows at drywell penetration X144, through wall cracks were identified on the exterior bellows ply. (During a recent Integrated Leak Rate Test (ILRT) at Quad Cities Nuclear Power Station, a significant leak had been discovered through the bellows of a similar drywell penetration. The ensuing inspections and investigations revealed that the normal LLRT methodology for these bellows was not representative of actual leakage.) As a result, the Dresden inspections were initiated. Six of the twenty four testable Unit 2 bellows were chosen for inspection. During the previous Unit 2 refuel outage, 8 of the 24 bellows possessed a positive leakage rate. The inspection lot of six was chosen from these eight based on accessibility and dose. Only the bellows at penetration X144 contained any indications of cracks.

The initial indications on bellows X144 were identified through a Penetrant Test (PT). Following the PT examination the bellows were pressurized and coated with a soap bubble solution to check for through leakage. This inspection revealed additional indications undetectable by PT. All the indications were then accurately documented on a baseline inspection map and sent to the Production Services Department for evaluation. The bellows were also inspected with helium leak detection equipment. The results of this inspection revealed leakage past both the inboard and outboard bellows plies.

C. CAUSE OF EVENT:

This report is being submitted in accordance with 10CFR 50.73(a)(2)(i)(B), which requires the reporting of any operation or condition prohibited by the Technical Specifications.

The unsatisfactory leakage from valve 2-1501-27A has been determined to be caused by an accumulation of foreign material in the valve seat. After flushing the valve seat with pressurized clean demineralized water, the leakage rate was significantly reduced. The cause of failure was initially suggested by the black mud like appearance of the water discharged from the volume of piping prior to the LLRT. Additionally, the fact that the previous two tests on this volume (performed on July 20, 1990 and November 22, 1988) yielded zero leakage further supports this conclusion.

A summary table of the root causes and corresponding corrective actions for these valves and the remaining type B and C tests which required repairs and adjustments is contained in section E of this report.

The root cause of the indications found on bellows X144 has been attributed to Transgranular Stress Corrosion Cracking (TGSCC). The bellows at Quad Cities station which was leaking was examined by the Systems Material Analysis Department (SMAD) metallurgists who concluded that the cracking mechanism was TGSCC. In addition, on two other occasions metallurgical examinations have been performed to evaluate bellows leakage. In each of these investigations the leakage was attributed to TGSCC. Thus, although the bellows at X144 has not yet been metallurgically examined, the evidence present suggests TGSCC.

D. SAFETY ANALYSIS OF EVENT:

The consequences of the type B and C test results described within this report are minimal because in all of the above mentioned failures (2-203-1A, 2-203-1D, 2-203-2C, 2-1501-27A, and all the valves and penetrations noted in the table in section B of this report) the in line isolation valves were observed to have satisfactory integrity. Upon completion of all the "as found" type B and C testing, the total "as found" minimum pathway (through) leakage was calculated to be 241.8 sfh which is well within the Technical Specification limit of 488 scfh (0.6 La).

The safety significance of the through wall cracks found on penetration X144 is also minimal. Sufficient margin exists in the results of the recent Unit 2 ILRT which was completed in December, 1990. The "as left" ILRT resulted in a Primary Containment leakage rate of 0.8128 weight %/day which provides a margin of 0.3782 weight %/day (199 scfh) from the Technical Specification limit of 1.2 weight %/day. It is believed that these cracks were present during the ILRT and thus accounted for accordingly in the "as left" results. In addition, an engineering analysis was performed to conservatively predict the potential leakage from the bellows at X144. This analysis assumed that in addition to the indications found on the bellows, two-hundred other subsurface flaws exist that would pinhole immediately after unit startup and grow to a length of 3/8" over the remainder of the operating cycle; and, that the existing 7/16" crack grows to a length of 5/8". The combined effect of these cracks was calculated to produce a leakage rate less than 1/2 the existing margin of the "as left" ILRT results.

It should be noted that performance of the type A ILRT in December, 1990 identified a significant leak from the inboard flange of valve Drywell to Torus Vacuum Breaker [BF] 2-1601-20A. This valve was tested during the outage as part of the Appendix J type C testing program; however, this test only challenges the valve seats and the outboard valve flange. Thus, the leak on the inboard flange was not found during the type C test. The incident involving the type A test failure was reported by LER #90-018 on Docket #050237.

E. CORRECTIVE ACTIONS:

The table below summarizes the root causes and the corresponding corrective actions performed on the MSIVs 2-203-1A, 1D, 2C, valve 2-1501-27A, and the remainder of the type B and C tests which required repairs and adjustments due to an excessive leakage rate.

Test Volume	Cause and Corrective Actions
2-1501-27A, 28A	The LPCI Containment Spray line containing these two valves was hydrolazed under the direction of WR D80092. The final LLRT was performed on December 20, 1991, resulting with a leakage rate of 0.1 SCFH.
2-203-1A, 2A	MSIV 2-203-1A was repaired under the direction of WR D95360. The inspection of the valve internals revealed cuts on seating area of the main disc. The valve internals were rebuilt utilizing a new main disc, pilot valve disc, pilot valve seat, valve upper guide liner, and new valve stem. The valve was reassembled and the final MSIV LLRT (dry) performed on December 16, 1990 yielding a leakage rate of 0.1 SCFH. A review of the LLRT history for this valves revealed that this was the first time it failed an LLRT.

Test Volume	Cause and Corrective Actions
2-203-1D, 2D	MSIV 2-203-1D was repaired under the direction of WR D95361. The inspection of the main valve internals found the pilot valve disc worn with an approximate 1/16" groove across the seating area. The valve internals were rebuilt utilizing a new pilot valve disc, upper guide liner, and valve stem. The valve was reassembled and the final MSIV LLRT (dry) performed on December 4, 1990 yielding a leakage rate of 4.0 SCFH. A review of the LLRT history for these valves revealed that the 2-203-1D failed during the previous refuel cycle due to wear on the pilot disc and seat. Prior to that, this valve had not failed since 1974.
2-203-1C, 2C	MSIV 2-203-2C was repaired under the direction of WR D95359. The inspection of the valve internals revealed steam cuts and wear on the main disc and seat. The valve internals were rebuilt utilizing a new main disc, pilot valve disc, and pilot valve seat. The main body seat ring was also lapped. The valve was reassembled and the final MSIV LLRT (wet) performed on December 9, 1990 yielding a leakage rate of 3.2 SCFH. A review of the LLRT history revealed that this valve has not failed an LLRT since 1983. An industry-wide data base search regarding leak rate testing failures of these type of MSIVs yielded 40 failures. Repairs included seating surface machining and packing adjustment/replacement.
2-205-2-7	Head Spray check valve 2-205-2-7 was repaired under the direction of WR D92531. Inspection of the valve internals revealed minor corrosion and debris on the valve seats. However, after the valve seats were cleaned, the valve still failed an LLRT bench test. The valve seating surfaces were then lapped which resulted in a successful bench test. The final LLRT was performed on December 10, 1990 yielding a satisfactory leakage rate of 7.9 SCFH. A review of the LLRT history revealed that this valve has not failed an LLRT since 1983 (refuel outage D2R8). An industry-wide data base search regarding this type of valve revealed no LLRT related failures.
2-220-57A, 58A	Feedwater check valve 2-220-58A was repaired under the direction of WR D95269. Inspection of the valve internals revealed no major problems with the disc-seat assembly. The body seat O-ring was found to be deteriorated but appeared to be functional. The disc also appeared to operate satisfactorily, although an as found blue check was not performed. The valve was reassembled using a new disc-seat assembly and new O-ring. The final LLRT was performed on December 4, 1990 yielding a satisfactory leakage rate of 1.8 SCFH. A review of the LLRT history revealed that this valve has not failed an LLRT since 1983 (refuel outage D2R8).
2-220-57B, 62B	Feedwater check valve 2-220-62B was repaired under the direction of WR D81758. The initial inspection revealed the disc stuck in the open position. Excessive force was required to close and open the valve disc. It is suspected that this binding was caused by hinge pin bushings which were not properly deburred. The station maintenance work package data file for feedwater check valve work has been updated to include a step ensuring that the bushings are properly deburred after line-boring. The valve was reassembled using a new disc-seat assembly and O-ring. The final LLRT was performed on November 26, 1990 yielding a satisfactory leakage rate of 4.9 SCFH. A review of the LLRT history revealed that this valve was found stuck open during the previous refuel outage (D2R11). The corrective actions involved replacing the disc-seat assembly. Prior to that, the valve had not failed since before 1979. An industry-wide data base search regarding this type of valve (identical to 2-220-58A) revealed 28 failures, 5 of which were LLRT related. Of the five LLRT failures, the root causes were worn out disc-seat assemblies and dirty valve internals. Corrective actions involved either replacing or lapping the disc-seat assembly.

Test Volume	Cause and Corrective Actions
2-301-160A, 161A	Preliminary inspections of Scram Discharge Volume Vent valve 2-301-161A suggested that the valve was not completely closing. Initial troubleshooting under the direction of WR D95902 involved adjusting the valve stroke length. The subsequent LLRT performed on November 3, 1990, yielded a satisfactory leakage rate of 10.3 SCFH. A review of the LLRT history for this valve revealed that this was its first failure. An industry-wide data base search regarding this type of valve revealed no additional failures
2-1001-1A,1B 2A, 2B, 2C	Shutdown Cooling Suction valve 2-1001-1A was repaired under the direction of WR D95639. The initial inspection revealed that the valve seats and wedge were washed out and pitted. The valve seats and a new wedge were lapped until a satisfactory blue check was achieved. The valve was reassembled with a live load packing arrangement and a rebuilt operator. Shutdown Cooling Suction valve 2-1001-1B was repaired under the direction of WR D95638. The valve seats and wedge were also washed out and pitted similar to the as found condition of valve 2-1001-1A. The valve seats and the original wedge were lapped until a satisfactory blue check was achieved. This valve was also reassembled with live load packing and a rebuilt operator. The final LLRT on this test volume was performed on December 9, 1990 yielding a leakage rate of 30.6 SCFH. A review of the LLRT history for these two valves revealed a previous failure in 1986 (refuel outage D2R10). The corrective actions for that failure involved lapping the valve seats and disc.
2-1301-3, 4	Isolation Condenser Return valve 2-1301-3 was repaired under the direction of WR D95268. Inspections of the valve internals revealed the valve seat and disc to be badly eroded. The disc was then stripped of its hardened surface, and then resurfaced and machined. The seats were then lapped to the disc. After achieving a satisfactory blue check, the valve was reassembled. The final LLRT was performed on December 15, 1990, yielding a satisfactory leakage rate of 0.1 SCFH. A review of the LLRT history for this valve revealed that it has not failed since prior to 1979. An industry-wide data base search for this valve and valves 2-1001-1A and 1B revealed two additional failures, although neither involved leakage past the valve seat.
2-1501-18B, 19B	During the "as found" LLRT on LPCI Torus Spray valves 2-1501-18B and 19B, an inspection of the vent upstream of valve 2-1501-18B revealed that this was the suspect valve. When the conditions were made available, the seating area of valve 2-1501-18B was flushed with pressurized clean demineralized water. The subsequent LLRT performed on November 2, 1990 yielded a satisfactory leakage rate of 7.2 SCFH. An industry-wide data base search for both valves, 2-1501-18B and 19B, revealed no failures involving this type of seat leakage.
2-2001-5, 6	Drywell Equipment Drain isolation valve 2-2001-6 was repaired under the direction of WR D96190. Inspections of the valve internals revealed no cuts or foreign material on the valve seats or disc. However, during the disassembly process it was noted that the valve packing was too tight. The disc and seat were then satisfactorily blue checked and the valve reassembled with new packing and a rebuilt air operator. After reassembly the valve was stroked several times to ensure no binding.

Test Volume	Cause and Corrective Actions
	Valve 2-2001-5 was repacked under the direction of WR D96191. During the initial LLRT leakage was noted from the valve packing. The final LLRT on this test volume was performed on December 21, 1990 yielding a leakage rate of 20.5 SCFH. A review of the LLRT history for these two valves revealed that they have not failed since prior to 1974. An industry-wide data base search regarding this type of valve revealed one additional failure where the valve was leaking condensate past the valve seat. Inspection of the valve revealed cuts on the seat and disc.
2-2301-34, 71	HPCI Drainpot to Torus valve 2-2301-71 was repaired under the direction of WR D95283. The valve internals were found to be very dirty and rusted. The valve stem was also badly pitted. The valve internals cleaned and the valve seats were lapped. After the valve was repacked and reassembled, a final LLRT was performed yielding a leakage rate of 5.1 SCFH. A review of the LLRT history for this valve revealed that it has not failed an LLRT since prior to 1974. An industry-wide data base search regarding this type of valve revealed an additional LLRT failure due to rust and dirt in the valve internals.
2-3702, 3799-126	RBCCW Supply to Drywell valve 2-3702 was repaired under the direction of WR D95414. The valve seats and disc were found to be rough and in need of lapping. The original disc and seats were lapped and a satisfactory blue check achieved. The valve was reassembled and subsequently tested yielding a leakage rate of 12.5 SCFH. This was the first time that this valve was correctly tested. An industry-wide data search revealed no additional failures.
Electrical Penetration 202BB	Initial troubleshooting into the source of the leakage involved removing the cover inboard and outboard sides of the penetration, pressurizing the penetration, and leak test (using soap bubble solution) the accessible areas. A major portion of the leakage was found coming from the Amphenol connectors and spare cable ends on the inboard side of the penetration. Under the direction of WR D95284, the ends of the spare cables which were leaking were capped using Raychem end seal caps. The subsequent LLRT however yielded a leakage rate of 24 SCFH, which was slightly greater than the initial test results. The leak rate for this penetration has been slowly trending up since 1981. An industry-wide data search revealed 1 LLRT related failure which was due to cracks in the ceramic face of the penetration. The cracks were sealed using an approved material. The Site Engineering department is currently investigating methods for repairing the leakage on this electrical penetration. This study is also investigating and evaluating the replacement of the penetration with a new type of electrical penetration. Upon completion of this study and after station discussion, the appropriate actions will be taken to reduce the leakage at this electrical penetration (237-200-90-10404).

Test Volume — Cause and Corrective Actions

Bellows X113 — Initial troubleshooting on bellows X113 revealed a pin hole leak on the test line running from the test station to the penetration. The section of the test line which contained the leak was then replaced under the direction of WR D95728. The subsequent LLRT however revealed only a slight decrease in the leak rate to 44 SCFH. A helium leak detection test was then performed per Special Procedure (SP) 91-1-11 to determine the location of the leakage. The containment side of the bellows was tested by pressurizing between the bellows plies with helium, and pulling an air sample from the top of the inside of the penetration pipe sleeve to the helium leak detector. The test results yielded no sign of leakage past the bellows inner ply. The test equipment was then set up to test the bellows outer ply. An air sample was pulled from underneath the top of the protective cover to the helium leak detector. This test yielded a positive indication of leakage past the outer ply. Arrangements are currently being made to replace this bellows during the next Unit 2 refueling outage (refuel outage D2R13) under the direction of Work Request D00671 (237-200-90-10401S2). An industry-wide data search regarding bellows of this type revealed 3 failures; however, all three were due to a leaking test valve.

Bellows X144 — Dresden Station has implemented a 5 step bellows testing program which will be used to test all two-ply bellows seals during each refuel outage. This program consists of the following steps:

1. Each two-ply bellows seal will be locally pressurized between the plies with air at a pressure not less than Pa to determine and quantify any leakage. If the leakage rate is less than 0.5 scfh, additional testing is not required.

2. If the leakage rate equals or exceeds 0.5 scfh, the bellows seal will be locally pressurized between the plies with helium at a pressure not less than Pa. The outer ply will then be tested for the presence of helium which would be an indication of leakage through the outer ply. Since both the inner and outer plies are qualified primary containment boundaries, additional testing is not required if no leakage is detected through the outer ply.

3. If helium leakage is detected through the outer ply, then an additional helium test will be performed and the inner ply will be tested for the presence of helium. If there is no indication of leakage past the inner ply, additional testing is not required since both the inner and outer plies are qualified primary containment boundaries.

4. If helium leakage is detected through both the inner and outer plies, then the protective guard will be removed and the bellows will be pressurized between the plies with air at a pressure not less than Pa. The surface of the outer ply will be tested with a soap bubble solution, and flaw indications will be measured, mapped, and evaluated.

Test Volume — Cause and Corrective Actions

Bellows X144
(Continued)

5. An Integrated Leak Rate Test will be performed prior to unit start-up to verify primary containment integrity.

Bellows seals which exhibit leakage through both plies will be replaced during the next refueling outage. Written technical justification must be provided to the NRC within ninety days after unit start-up if this replacement schedule is not implemented for any subject bellows. This testing program for two-ply bellows will remain in effect until a bellows is replaced with a new testable type or until a valid Type B test can be employed.

Unit 2 bellows penetration X-144 was tested with soap-bubble solution in April, 1991. The surface flaws on the outer ply were measured, mapped, and evaluated. The Commonwealth Edison Nuclear Engineering Department determined that the bellows were operable and would remain operable for at least one cycle. Additional testing of the inner ply of the X-144 bellows was performed in January, 1992. At this time, a helium test was performed to identify leakage past the inner ply. The results of this test indicated that there was leakage through the inner ply. Long term corrective actions for this penetration include the following: During the upcoming D2R13 refuel outage, Modification M12-2-91-003 will be performed to cut and cap the Control Rod Drive Return Line (CRDRL) which penetrates penetration X-144. In addition, the modification will cap the penetration inside of containment, thereby eliminating the bellows as part of the primary containment boundary.

Since the full intent of the new bellows test program has been satisfied for penetration X-144, a mid cycle inspection, as given to in LER #90-009-1, does not need to be performed. The bellows testing program, as described in items 1 through 5 above, will be procedurally controlled via Dresden Technical Staff Surveillance (DTS) 1600-02, LLRT of Bellows Assemblies. DTS 1600-02 will be revised by the Technical Staff prior to the start of the next Dresden refuel outage (D2R13) to reference these requirements (237-200-90-10402S2).

F. PREVIOUS OCCURRENCES:

LER/Docket Numbers	Title
89-009/050249	Local Leak Rate Testing "As Found" Limit Exceeded Due to Excessive Leakage From Primary Containment Isolation Valves

A major contributor to this event was leakage from reactor building [NG] to pressure suppression chamber vacuum breaker check valve 3-1601-31B (redundant isolation valve not leaking). The cause of this failure was attributed to worn hinge pin which caused the valve seat to bind prior to it reaching the fully closed position.

G. COMPONENT FAILURE DATA:

Manufacturer	Nomenclature	Model Number	MFG. Part Number
Crane Co.	Inboard MSIVs 2-203-1A, 1D, 2C	DR34289-20" Y Pattern Globe Valve	N/A
Mission Mfg.	Head Spray Check Valve 2-205-2-7	90 CPF-403	N/A
Crane Valve Co.	Feedwater Check Valve 2-220-58A and 62B	973	N/A
Crane Valve Co.	Shutdown Cooling Suction Valves 2-1001-1A, 1B, and Isolation Condenser Return Valve 2-1301-3	783-UL	N/A
Crane Valve Co.	LPCI Torus Spray Valves 2-1501-18B, and 19B	151-1/2XR/33-1/2U	N/A
Crane Valve Co.	Drywell Equipment Drain Isolation Valve 2-2001-5, and 6	47-1/2XR	N/A
Henry Vogt Machine Co.	Scram Discharge Volume Vent Valve 2-0301-161A	15111-SW	N/A
Rockwell International	HPCI Drainpot to Torus Valve 2-2301-71	868Y	N/A
Crane Valve Co.	RBCCW Supply Valve 2-3702 Valve 2-3702	47-1/2U	N/A
General Electric	Electric Penetration X202BB	N/A	N/A
Pathway Bellows	RWCU Penetration X113	N/A	N/A
Pathway Bellows	CRD Return Penetration X144	N/A	N/A

An industry-wide data base search regarding leak rate testing failures was performed for the above mentioned valves and penetrations. The information obtained from this search is contained within the applicable sections of Section E of this report.

ATTACHMENT 1

MSIV LEAKAGE RATES

Main Steam Line	"Dry" Test Results (SCFH)	"Wet" Test Results (SCFH)	Outboard MSIV Leakage Rates (2-203-2A,B,C and D) (SCFH)	Inboard MSIV Leakage Rates (2-203-2A, 1B, 1C 1D) (SCFH)
A	33.0	9.5	9.5	23.5
B	6.7	4.0	4.0	2.7
C	17.8	14.2	14.2	3.6
D	39.0	Undetermined*	--*	39.0*

* Unable to properly perform "Wet" test due to excessive water inleakage to the test volume through the inboard MSIV. Consequently, it was believed that the 39.0 SCFH leakage rate was due to the condition of the inboard MSIV; this was confirmed by performing further testing upon completion of repairs.