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ComEd

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JSP Ltr. #96-0043

U.S. Nuclear Regulatory Commission
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Enclosed is Licensee Event Report 95-011, Docket 50-249, Revision 2, which is being submitted pursuant to 10CFR50.73(a)(2)(i)(b), which requires the reporting of any operation or condition prohibited by the plant's Technical Specifications. This Supplemental LER is provided to provide additional dose rate information pertaining to the event.

Sincerely,

for
J. Stephen Perry
Vice President
BWR Operations

Enclosure

010113

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

JE22

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 3	05000249	1 OF 7

TITLE (4)
Type B and C Leakage Limit Exceeded Due to Ineffective Corrective Actions for Past Valve Failures

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 None	DOCKET NUMBER
05	29	95	95	-- 011 --	02	03	26	96	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	N	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)
POWER LEVEL (10)	000		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 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	BO	ISV	C684	Yes
X	BO	ISV	C684	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 1600 on May 29, 1995, with Unit 3 shutdown for maintenance, the performance of Dresden Technical Surveillance (DTS) 1600-01, Local Leak Rate Testing Of Primary Containment Isolation Valves, identified the High Pressure Coolant Injection (HPCI) System Turbine Exhaust to Suppression Pool Check Valve 3-2301-45 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 Technical Specification maximum pathway leakage rate limit for Type B and C primary containment leakage was exceeded. The safety significance of the leakage past the 3-2301-45 was considered to be minimal since the additional minimum pathway leakage out of containment was 14.39 scfh from the inboard isolation Stop Check Valve 3-2301-74 (LLRT performed on 9/2/94) 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 start-up. This supplement is submitted to provide additional dose rate information.

EVENT IDENTIFICATION:

Type B and C Leakage Limit Exceeded Due to Ineffective Corrective Actions for Past Valve Failures

A. PLANT CONDITIONS PRIOR TO EVENT:

Unit: 3	Event Date: 05/29/95	Event Time: 1600 hrs.
Reactor Mode: N	Mode Name: Shutdown	Power Level: 0%

Reactor Coolant System Pressure: 0 psig

B. DESCRIPTION OF EVENT:

At approximately 1600, on May 29, 1995 with Unit 3 shutdown for maintenance, the performance of Dresden Technical Surveillance (DTS) 1600-01, Local Leak Rate Testing Of Primary Containment Isolation Valves, identified the High Pressure Coolant Injection (HPCI) System [BJ] Turbine Exhaust to Suppression Pool Check Valve 3-2301-45 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.

To verify that corrective actions previously undertaken to prevent HPCI check valve LLRT failures were sufficient, leak rate tests were to be performed on an accelerated schedule of every six months. This first six month leak rate test resulted in an LLRT failure of the 3-2301-45 valve.

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.

The 3-2301-45 valve was replaced after re-installing the original-installed closing springs (109 in.-lbs). The as-left Local Leak Rate Test yielded a leakage rate of 0.10 scfh (NTS #249-180-95-01101). Testing with acoustics and pressure sensors was performed on the 3-2301-45 valve to determine the actual condition during the start-up HPCI runs after Maintenance Outage D3F18. Monitoring will also be performed during the next HPCI run. Evaluation of the results will determine future actions.

Unrelated local leak rate testing identified two additional valves that were leaking more than the test equipment could measure. The Low Pressure Coolant Injection (LPCI) System [BO] Loop I Drywell Spray Outboard Isolation Valve 3-1501-27A and Inboard Isolation Valve 3-1501-28A were found to have an indeterminate amount of leakage.

The 3-1501-27A valve was disassembled and inspected. Piping corrosion products were found to have scratched the valve disk and seating surfaces. These seating surfaces were cleaned and polished. Local leak rate testing of the spray volume still yielded a leakage of approximately 300 scfh. The 3-1501-28A valve was then disassembled and inspected. Piping corrosion products were found to have scratched the valve disk and seats. These seating surfaces were also cleaned and polished. An as-left LLRT yielded a satisfactory leakage rate of 0.10 scfh.

Valve 2301-45 reliability was reviewed and approved by the Plant Operations Review Committee on August 31, 1995 (NTS #249-180-95-01103)

C. CAUSE 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.

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 3-2301-45 valve was removed from the system, the inspection revealed the seats to be in good condition. The inspection also indicated that the disks would not go fully closed. As the disks were released from the full open position, the disks would stop midcycle leaving a gap as large as one inch between the seat and the disks. This gap is large enough so that flow through a 3/4" LLRT test tap would pass through the gap and not provide adequate force to fully close the disks 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 disks 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.).

Therefore, the root cause of the failure of the 3-2301-45 valve on May 29, 1995, was ineffective corrective actions implemented from a previous LLRT failure.

The cause of the scratched seating surfaces of the LPCI Drywell Spray Valves 3-1501-27A and 3-1501-28A was due to piping corrosion products present on seating surfaces when the valve is stroked. Operating Department procedures keep the piping volume between the spray valves drained of water. This condition is leading to increased corrosion of the piping and subsequently these corrosion products are ending up on valve seating surfaces. These Crane solid wedge gate valves have a seating surface material which is more susceptible to being scratched than stellite. In addition, the valve team has increased the torque setting of the valves by a factor of two in response to Generic Letter 89-10 concerns. Increased corrosion products on seating surfaces when the valve is stroked with a greater force has led the scratched seating surfaces.

D. SAFETY ANALYSIS:

The safety significance of the leakage past the 3-2301-45 was considered to be minimal since the minimum pathway leakage through this containment penetration was 14.39 scfh past the inboard isolation Stop Check Valve 3-2301-74 (LLRT performed on 9/2/94) and would not have caused the maximum off-site dose rates established in 10 CFR 100 to be exceeded.

The safety significance of the leakage past the 3-1501-27A and 3-1501-28A was considered to be minimal since the minimum pathway leakage through the containment penetration was at most 300 scfh past the inboard isolation valve 3-1501-28A and would not have caused the maximum off-site dose rates established in 10 CFR 100 to be exceeded.

Additional Unit 3 Primary Containment Isolation Valves experienced leakage, therefore, a review of total containment leakage was performed. This leakage included leakage from the containment liner, containment head, downcomer, suppression pool, welds, valves, penetrations, piping and instrumentation. Two of these valves were Anchor Darling dual disk gate valves installed in the Main Steam Line Drains (MSLD) (Ref. LER/Docket Number 95-007/0500249). Assuming the worst case break of the Main Steam Line Drain pipe in the Turbine Building, preliminary calculations using Station Off-site Dose Calculation Manual software for Control Room Operator dose, Exclusion Area Boundary (EAB) dose, and Low Population Zone (LPZ) dose were performed based on minimum pathway leakage from the Dresden Unit 3 Primary Containment. Under worst case conditions, dose limits for Control Room Operators established by General Design Criteria (GDC) 19 of Appendix A to 10 CFR 50 as well as EAB and LPZ dose limits established by 10 CFR 100 would have been exceeded during the Design Basis Accident.

During a LOCA DBA the control room HVAC Air Filtration Unit (AFU) will help reduce radiation exposures within the Control Room. The AFU provides radiation protection by pressurizing the control room emergency zone with 2000 cfm of filtered make up air from outside containment. The AFU maintains the CR emergency zone at +1/8 inch water pressure post LOCA so that the radiation exposure to the Control Room personnel would be reduced. Bottled air and iodine tablets are also available to the Operators to mitigate the effects of airborne contamination in the control room.

If a loss of off site power (LOOP) would have occurred at the same time as the LOCA, the HVAC mitigating factor relative to radiation exposure to control room personnel would not have been available. However, if a coincident LOOP and LOCA were to occur, other factors would have been in place to mitigate against control room radiation levels. The control room HVAC automatically isolates which maintains the same air within the control room. Thus, the air already in the control room would be free of radionuclides released during a LOCA except for the amount brought in through inleakage. In addition, the standby gas treatment system (SBGTS) charcoal efficiency is higher than the 90% as required by the technical specifications. This would provide for some removal of radionuclides from the reactor building. Also, the control room does not have any exterior walls affected by outside wind which could affect infiltration into the control room.

As documented in Corporate Nuclear Fuel Services assessment NFS:BSA:95-110, Radiological Assessment of Leakage Through MS Drain Line, the conservative design-basis radiological dose assessments are performed assuming that the post-LOCA containment leakage is the Technical Specification leakage rate, 1.6 wt% per day. This performance is demonstrated by periodic leakage rate testing at design pressure, 48 psig. The most recent Integrate Leak Rate test, combined with local leak rate tests, indicates the present actual containment leakage rate into the reactor building (secondary containment) is 1.27 wt% per day, or about 20% lower than the design basis value. The containment releases would be expected to be lower by the same proportion. In addition, the containment pressure is not sustained at 48 psig over the course of the accident, but decreases due to containment sprays and other Engineered Safety Features.

A realistic assessment of the radiological consequences of a LOCA will result in CR and offsite doses that are lower than those calculated using regulatory guidance. First, consideration of the expected pressure profile will result in 52% lower releases; consideration of the actual, as-tested containment leakage rate to the reactor building reduces the leakage by an additional 20%. Second, consideration of the timing of releases from the reactor core will remove releases concurrent with containment peak pressure (and high release rate to the environment) from concern. The delayed beginning of the release impacts mainly the 2 hour LPZ dose. Third, because the chemical form of iodines is such that it remains in solution and does not become airborne, the environmental impact of iodines will be reduced by at least an additional factor of 10 as compared to the design basis scenario.

A realistic estimate of the post-LOCA radiological dose via the MSL drain line pathway may be estimated from the conservative design basis dose values by scaling from the leakage rates for the two pathways. The leakage through the MSL drain line bypasses secondary containment and thus loses the benefit provided by the SBGT. However this is countered by the reduction in post-LOCA airborne radioiodine inventory offered by the soluble chemical form now considered.

When the realistic MSL drain leakage values are combined with the realistic containment leakage component of the post-LOCA releases, the thyroid and whole body doses are less than the applicable 10CFR100 and 10CFR50 (GDC 19) limits.

This event is potentially significant due to the possible release of radioactivity to the public due to total containment leakage, however, the safety significance of the leakage from Primary Containment is considered to be minimal because there was no challenge to containment and no release.

E. CORRECTIVE ACTIONS:

Monitoring of the High Pressure Coolant Injection System Turbine Exhaust to Suppression Pool Check Valve, 3-2301-45, will also be performed during the next HPCI run. Evaluation of the results will determine if additional actions are required. (NTS #249-180-95-01104)

Local Leak Rate Testing of the 3-2301-45 valve will continue to be performed on an accelerated interval. An LLRT is performed after six months if the unit were to shutdown or a HPCI Limiting Condition for Operation (LCO) of greater than three days is entered for any reason other than differential pressure testing.

Other utilities were contacted in an effort to investigate the possibility of a new design for this check valve. Utilities with a lift-type check valve in this application experienced the least amount of failures. Replacing the current dual disk check valve, 3-2301-45, with a lift-type check valve is being evaluated. (NTS #249-180-95-01105)

Evaluation of valve design for the Low Pressure Coolant Injection System Drywell Spray Outboard (3-1501-27A) and Inboard (3-1501-28A) valve is ongoing. (NTS #249-180-95-01106)

A review of system operation coupled with procedure changes will be performed to ensure that the piping between valves 2(3)-1501-27A(B) and 2(3)-1501-28A(B) remains water filled to minimize pipe corrosion and to prevent corrosion products from coming in contact with the seating surfaces. (NTS #249-180-95-01107)

F. PREVIOUS OCCURRENCES:

LER/Docket Numbers	Title
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
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 seventeen corrective maintenance entries for C & S Valve Co. check valves. Six failures were due to corrosion of valve internals and ten failures were due to worn seating surfaces.

Manufacturer	Nomenclature	Model Number	Mfg. Part Number
Crane Valve Co.	LPCI Loop I Drywell Spray Outboard Isolation Valve 3-1501-27A	33-1/2XR	N/A
Crane Valve Co.	LPCI Loop I Drywell Spray Inboard Isolation Valve 3-1501-28A	33-1/2XR	N/A

An industry-wide search revealed four corrective maintenance entries for Crane Valve Co. valves. None of the failures were due to scratches on valve seats.