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ComEd

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U.S. Nuclear Regulatory Commission
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Enclosed is Licensee Event Report 95-007, 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 supplement is being submitted to provide additional information concerning the dose rates, root cause, and corrective actions pertaining to the event.

Sincerely,

[signature] FOR
J. Stephen Perry
Vice President
BWR Operations

Enclosure

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

010114

JE22

NRC FORM 366
(5-92)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

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.

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Dresden Nuclear Power Station, Unit 3	05000249	1 OF 7

TITLE (4)
Leakage Limit Exceeded Due to Valve Internal Damage Caused by Manual Operation of MOV

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
06	30	95	95	-- 007 --	02	03	26	96	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 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	SB	ISV	A391	Yes						
X	CE	ISV	A391	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 2130, on June 30, 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 Main Steam Line Drain (MSLD) gate valves 3-220-1 and 3-220-2 to be leaking more than the test equipment could measure. Closing of the 3-220-2 valve by means of manual engagement of the Motor Operated Valve's (MOV) handwheel resulted in valve internal damage and a leakage path. The causes of manual operation resulting in damage to the valve was that the design review of the modification to change the style and manufacture of the valve did not identify the low torque value (approximately 33 foot-pounds) that would damage the valve during manual operation. Contributing cause to the 3-220-2 valve failure was the informally controlled verification of OOS placements on MOV handwheels. The failure of the 3-220-1 and 3-1201-1 valves was due to improper fit up. The improper fit up was due to human error during the fit up process following the valve replacement during D3R13. The corrective actions included repairing or replacing the valves and procedure revisions to improve MOV out of service controls. This supplement is being submitted to provide additional information concerning the dose rates, root cause, and corrective actions.

EVENT IDENTIFICATION:

Leakage Limit Exceeded Due to Valve Internal Damage Caused by Manual Operation of MOV

A. PLANT CONDITIONS PRIOR TO EVENT:

Unit: 3 Event Date: 06/30/95 Event Time: 2130

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

Reactor Coolant System Pressure: 0 psig

B. DESCRIPTION OF EVENT:

During Refuel Outage D3R13 (March 1994 - November 1994), the Main Steam Line Drain (MSLD) [SB] Primary Containment Isolation Valves 3-220-1 and 3-220-2, Crane gate valves, were cut out and replaced with Anchor Darling dual disk gate valves.

On January 16, 1995, with Unit 3 exiting Maintenance Outage D3F17, the MSLD inboard gate valve 3-220-1 was given a close signal. Dual indication (both open and closed), not a full close indication, was received in the Control Room. The valve was opened and again given a close signal. This time the Control Room received a full close indication. Since the indication was erratic, this Primary Containment Isolation Valve was declared inoperable.

Technical Specification 3.7.D.1. states:

> During reactor power operation conditions, all primary containment isolation valves and all instrument line flow check valves shall be operable except as specified in 3.7.D.2.

Technical Specification 3.7.D.2. states:

> In the event any primary containment isolation valve becomes inoperable, reactor power may continue provided at least one valve in each line having an inoperable valve is in the mode corresponding to the isolated condition.

Therefore, the MSLD outboard gate valve 3-220-2 was taken Out-of-Service in the closed position. This stopped the clock for the Limiting Condition for Operation described in Technical Specification 3.7.D.3. which states:

> If specification 3.7.D.1 and 3.7.D.2 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

When Dresden Unit 3 shutdown for Maintenance Outage D3F18, the Motor Operated Valve (MOV) team began investigating the valve's position discrepancy. Limits were found to be engaged and the valve appeared to be closed. The torque switch setting was then increased in order to increase margin between the minimum required thrust and thrust developed at the torque switch setting. At approximately 2100, on June 14, 1995, the performance of Dresden Technical

Surveillance (DTS) 1600-01, Local Leak Rate Testing Of Primary Containment Isolation Valves, identified the MSLD gate valves 3-220-1 and 3-220-2 to be leaking more than the test equipment could measure. Checking of the vent path revealed that the MSLD outboard gate valve 3-220-2 was leaking greatly. The Unit Supervisor was notified of the event and a Performance Improvement Form (PIF) was written. The ComEd Reportability Manual states:

> In general, for the purpose of evaluating the reportability of situations found during surveillance tests, it should be assumed that the situation occurred at the time of discovery, unless there is firm evidence to believe otherwise.

On May 29, 1995, the High Pressure Coolant Injection (HPCI) System [SJ] check valve 3-2301-45 had leaked great enough to cause the cumulative Type B and C leakage to exceed the Technical Specification leakage limit of 0.60 L_a (488.452 standard cubic feet per hour). This failure was reported in LER 95-011/ Docket 50249. Reporting of the MSLD test volume LLRT failure was to be included in the supplement to that LER. However, the significance of the inboard valve failure described below required this report to be submitted.

On June 16, 1995, the MSLD test volume was pressurized and the outboard gate valve 3-220-2 was manually opened and then closed while the vent path was monitored. The leakage past the 3-220-2 was still unmeasurable. Valve disassembly revealed valve internal damage, a bent valve stem and valve internals missing. The Unit Supervisor was notified and a PIF (2492009506300) was written. An inspection of the piping with a boroscope did not locate the missing valve parts. Due to the internal damage suffered by the valve, the valve was cut out. To verify primary containment integrity, a plug was installed in the MSLD piping in order to Local Leak Rate Test the inboard gate valve 3-220-1.

At approximately 2130, on June 30, 1995, the performance of Dresden Technical Surveillance (DTS) 1600-01, Local Leak Rate Testing Of Primary Containment Isolation Valves, identified the Main Steam Line Drain (MSLD) gate valve 3-220-1 to be leaking more than the test equipment could measure.

The Unit Supervisor was notified of the event, and an ENS phone notification was then made at 0400 Eastern Standard Time on Saturday July 1, 1995 to report a condition that was outside of the design basis of the plant and a PIF was written to report a condition prohibited by the plant's Technical Specifications.

The MSLD test volume was again pressurized and the inboard gate valve 3-220-1 was manually opened and then closed while the leakage rate was monitored. The valve was left in the position where leakage was at its lowest rate, 17 scfh (standard cubic feet per hour).

A contact check of the disk and seating surfaces of the 3-220-1 valve determined that there were low spots on each side of the valve's outboard seat. Dresden Mechanical Procedure (DMP) 0040-58, Anchor Darling Dual Disk Gate Valve Maintenance, recently written to provide proper instructions for fit up of the valve disks, was used to correctly fit up the 3-220-1 and 3-220-2 valves after the performance of repairs. An as-left LLRT yielded a 0.10 scfh leakage rate.

Due to the two MSLD valve failures coupled with knowledge of recent problems with this type of valve at other stations, Engineering determined that Local Leak Rate Testing of the other Anchor Darling dual disk gate valves was warranted. Other Crane gate valves which were replaced with Anchor Darling dual disk gate valves during Refuel Outage D3R13 are the Reactor Water Cleanup (RWCU) System [CE] suction valves 3-1201-1, 3-1201-1A and 3-1201-2 and the Reactor Head Spray System [BO] valve 3-205-24. These systems were taken Out-of-Service in order to be given an LLRT. The LLRT on the RWCU valves yielded a leakage which was too great to be measured by the test equipment. Trouble shooting determined that the excessive leakage was past the inboard valve 3-1201-1 and its bypass valve 3-1201-1A, which are in parallel, and that the outboard valve 3-1201-2 was leaking 5 scfh. The LLRT of the Reactor Head Spray System gate valve 3-205-24 yielded a leakage of 1 scfh.

Light lapping was performed on the 3-1201-1A valve as the seat to disk revealed good contact. After disassembly of the 3-1201-1 valve, a check of seat to disk contact revealed a low spot in the seating surface. The 3-1201-1 valve had the seating surfaces lapped to repair the low spots on the valve's outboard disk and DMP 0040-58 was performed to ensure correct fit up. During the post-maintenance LLRT, the 3-1201-1 valve, in parallel with the 3-1201-1A valve, showed a leakage of 26 scfh which is higher than the administrative limit of 20 scfh. It was determined that due to radiological concerns it would be better to improve the leak rate during D3R14, since the impact to the overall containment leakage was minimal.

C. CAUSE OF EVENT:

This LER is being submitted pursuant to 10 CFR 50.73(a)(2)(i)(B) 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.

Disassembly of valves 3-220-1 and 3-1201-1 determined improper fit up of the seating surfaces to be the failure mode of the leakage. The improper fit up was most likely due to human error during the fit up process following the valve replacement during D3R13, contributing to this was a lack of procedural guidance and a lack of experience with Anchor Darling valve maintenance since these are new valves, not installed anywhere else in the plant.

It was determined that the failure mode of the damage to the 3-220-2 valve was excessive thrust being applied to the valve. This occurred during manual operation of the MOV handwheel in accordance with DOS 0201-02 (rev 14), Unit 3 RPV ASME B & PV Code 1000 PSI System Leakage Test/ Hydrostatic Test following D3R13 and when placing an out of service (OOS) card following D3F18.

The root cause of the failure on the 3-220-2 valve was that the design review of the modification to change the style and manufacture of the valve during June of 1994 did not identify the low torque value (approximately 33 foot-pounds) that would damage the valve during manual operation.

Contributing cause to the 3-220-2 valve failure was the informally controlled verification of OOS placements on MOV handwheels.

D. SAFETY ANALYSIS:

The safety significance of the leakage past the 3-1201-1 was considered to be minimal since the minimum pathway leakage through this containment penetration was 5 scfh past the outboard isolation valves 3-1201-2 and 3-1201-3 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 to determine the safety significance of the total containment leakage. This leakage included leakage from the containment liner, containment head, downcomer, suppression pool, welds, valves, penetrations, piping and instrumentation. Assuming the worst case break of 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:

The 3-220-1, 3-1201-1, and 3-1201-1A valves were repaired and the 3-220-2 valve was replaced.

Dresden Mechanical Procedure (DMP) 0040-58, Anchor Darling Dual Disk Gate Valve Maintenance, is in use to provide proper instructions for fit up of valve disks to seats.

Dresden Administrative Procedure (DAP) 03-05, Out-of-Service and Personnel Protection Cards, was revised to formally control handwheel usage during out-of-service placements and to note the potential to invalidate LLRTs when the MOV is operated manually.

DAP 07-27, Independent Verification, was revised to include the proper technique for verifying a MOV closed to prevent unnecessary handwheel operation.

The responsible Design Engineer and package reviewers have been counseled regarding the importance of identifying changes to the operating parameters/ modes when a modification is designed/ reviewed. They have learned from this event and have amended their work practices to prevent future occurrence.

The Engineering and Support Personnel continuing Training (ESPT) included this issue during a training session. The class examined this event, as well as other related events, and discussed how those events could have been prevented or mitigated during the design/ review process.

Engineering determined which safety related MOVs are susceptible to damage from handwheel usage, for units 2 and 3, and verified the valves' integrity.

Engineering developed a listing of the allowable generic torque values for safety related and balance of plant MOVs. These values will be used when the valves must be operated manually.

The Mechanical, Electrical, and Instrumentation departments reviewed the process for verifying MOVs OOS closed during a tailgate sessions. The message conveyed was that valves which are OOS cannot be verified closed by using the handwheel. If the crew member doubts that the valve is closed, then Operations would have to be contacted to perform the verification.

Non-Licensed Operators' continuing training, cycle 5, has been updated to address the hanging of OOS card(s) on MOV handwheels.

DOS 0201-02 was revised to assure any required manual operation of MOVs is consistent with station procedure for MOV manual operation.

F. PREVIOUS OCCURRENCES:

<u>LER/Docket Numbers</u> <u>Title</u>

None associated with Anchor Darling dual disk gate valves.

G. COMPONENT FAILURE DATA:

Manufacturer	Nomenclature	Model Number	Mfg. Part Number
Anchor Darling Valve Co.	MSLD 3-220-1 3-220-2	DD	N/A
	RWCU 3-1201-1 3-1201-1A	DD	N/A

An industry - wide data base search revealed 148 corrective maintenance entries for the Anchor Darling Model DD dual disk gate valve. Five failures were attributed to internal valve damage or misalignment of valve internals.