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Dresden Generating Station
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ComEd

July 5, 1995

TPJLTR 95-0074

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D. C. 20555

Licensee Event Report 95-015, Docket 50-237 is being submitted as required by Technical Specification 6.6 and 10CFR50.73(a)(2)(i) and 10CFR50.73(a)(2)(ii).

Sincerely,

T.P. Joyce

Thomas P. Joyce
Site Vice President

TPJ/:pt

Enclosure

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

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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 2	05000237	1 OF 4

TITLE (4)
Leakage Limit Exceeded Due to Excessive Leakage Past Main Steam 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 NAME	DOCKET NUMBER
									None	
06	07	95	95	-- 015 --	00	07	07	95	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	SB	ISV	C665	Yes						

SUPPLEMENTAL REPORT EXPECTED (14)				EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
X	YES (If yes, complete EXPECTED SUBMISSION DATE).		NO		11	13	95

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

At approximately 0215, on June 7, 1995, with Unit 2 shutdown for Refuel Outage D2R14, the performance of Dresden Technical Surveillance (DTS) 0250-01, Main Steam Isolation Valve Local Leak Rate (Dry) Test, identified the A and C Main Steam Isolation Valves (MSIVs) leaking 33.1 and 18.8 standard cubic feet per hour (scfh) respectively. On June 8, 1995, the performance of DTS 0250-03, Main Steam Isolation Valve Local Leak Rate (Wet) Test, identified the inboard A MSIV to be leaking 33.1 scfh and the inboard C MSIV to be leaking 14.85 scfh. These leakage rates exceed the limit specified in Technical Specification 3.7.A.2.b.2.c, which limits the leakage past any MSIV to 11.5 scfh when tested with air at a pressure of 25 psig. The safety significance of the leakage past the inboard A and C MSIVs was considered to be minimal since the additional leakage out of containment, on a minimum pathway basis, was 0 and 3.95 scfh, respectively, from the outboard MSIVs and would not cause the maximum off-site dose rates established in 10 CFR 100 to be exceeded. These valves will be inspected, repaired and Local Leak Rate Tested prior to unit startup. A supplement will be submitted to report the reason for valve failure and the corrective actions taken.

EVENT IDENTIFICATION:

Leakage Limit Exceeded Due to Excessive Leakage Past Main Steam Isolation Valves

A. PLANT CONDITIONS PRIOR TO EVENT:

Unit: 2 Event Date: 06/07/95 Event Time: 0215 hrs.

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

Reactor Coolant System Pressure: 0 psig

B. DESCRIPTION OF EVENT:

At approximately 0215, on June 7, 1995, with Unit 2 shutdown for Refuel Outage D2R14, the performance of Dresden Technical Surveillance (DTS) 0250-01, Main Steam Isolation Valve Local Leak Rate (Dry) Test, identified the A and C Main Steam Isolation Valves (MSIVs) [SB] leaking 33.1 and 18.8 standard cubic feet per hour (scfh) respectively. DTS 0250-01 is performed by pressurizing between the two MSIVs on each main steam line and thus finding the leakage through both valves.

The Unit Supervisor was notified of the excessive leakage and a Performance Improvement Form was initiated. Additional trouble shooting was performed to determine whether an individual MSIV exceeded its Technical Specification leakage limit.

This trouble shooting test is performed by flooding the Reactor Vessel and thus the main steam lines up to the inboard MSIV. Since the pressure exerted by the head of water is more than the LLRT test pressure, leakage found during the test is attributed to the outboard MSIV. When the leakage from the outboard MSIV is subtracted from leakage from both valves, the result is the leakage attributed to the inboard MSIV. On June 8, 1995, the performance of DTS 0250-03, Main Steam Isolation Valve Local Leak Rate (Wet) Test, identified the outboard A MSIV (2-0203-2A) to be leaking 0 scfh and the outboard C MSIV (2-0203-2C) to be leaking 3.95 scfh. Thus the inboard A MSIV (2-0203-1A) was leaking 33.1 scfh and the inboard C MSIV (2-0203-1C) was leaking 14.85 scfh. These leakage rates exceed the limit specified in Technical Specification 3.7.A.2.b.2.c, which limits the leakage past any MSIV to 11.5 scfh when tested with air at a pressure of 25 psig.

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.

These valves will be inspected, repaired and Local Leak Rate Tested prior to unit startup.

A supplement to this LER will be submitted to document the cause of the 1A and 1C MSIV LLRT failures.

D. SAFETY ANALYSIS:

The safety significance of the leakage past the inboard A and C MSIVs was considered to be minimal since the additional leakage out of containment, on a minimum pathway basis, was 0 and 3.95 scfh, respectively, from the outboard MSIVs and would not cause the maximum off-site dose rates established in 10 CFR 100 to be exceeded.

E. CORRECTIVE ACTIONS:

Nuclear Tracking System (NTS) tracking code numbers are identified in the text as (XXX-XXX-XX-XXXXX).

The 1A and 1C Main Steam Isolation Valves will be inspected, repaired and Local Leak Rate Tested prior to unit start up. (NTS #237-180-95-01501)

An LER supplement will be submitted which contains the cause of the MSIV LLRT failures, the repairs performed, and the results of the as-left LLRTs. (NTS #237-180-95-01502)

F. PREVIOUS OCCURRENCES:

LER/Docket Numbers	Title
93-026/0500237	Main Steam Line Isolation Valves 2-203-2A and 2-203-1D As-Found Leakage Rates Exceeded the Technical Specification Limit of 11.5 scfh
93-003/0500237	Outboard Main Steam Line Isolation Valve 2-203-2A As Found Leakage Rate Exceeded the Technical Specification Limit of 11.5 scfh
90-009/0500237	Type B and C Primary Containment Local Leak Rate Test Requirements Exceeded Due to leaking Isolation Valves
88-018/0500237	Leak Rate Limits Exceeded in Drywell Head Seal and MSIV 2-203-1D Tests Due to Misalignment and Seat Wear

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
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TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

G. COMPONENT FAILURE DATA:

Manufacturer	Nomenclature	Model Number	Mfg. Part Number
Crane Co.	Main Steam Isol. Valve 2-0203-1A	DR34289-20" Y Pattern Globe Valve	N/A
Crane Co.	Main Steam Isol. Valve 2-0203-1C	DR34289-20" Y Pattern Globe Valve	N/A

An LER supplement will be submitted with the results of an industry wide Nuclear Plant Reliability Data System (NPRDS) data base search of similar valve failures.