NRC Form 366
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/85

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Dresden Nuclear Power Station - Unit 2	0 5 0 0 0 2 3 7	1 OF 0 2

TITLE (4) Leakage in Excess of Technical Specification 3.7.A.2.b Limit Found on Main Steam Isolation Valve 2-203-2B Due to Packing Leak

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 5 0 0 0
1 2	1 4	8 6	8 6	0 3 0	0 0	0 1	0 6	8 7	N/A	0 5 0 0 0

OPERATING MODE (9)	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (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)	30.36(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 in Abstract below and in Text, NRC Form 366A)
	20.405(a)(1)(iii)	X 50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	
	20.405(a)(1)(iv)	50.73(a)(2)(ii)	50.73(a)(2)(viii)(B)	
	20.405(a)(1)(v)	50.73(a)(2)(iii)	50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER
Brian C. McCabe Technical Staff Engineer (X-483)	AREA CODE 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	MANUFACTURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	
X	S B	I S N	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 December 14, 1986 with Unit 2 at 0% power and in a refueling outage, main steam isolation valve (MSIV) 2-203-2B was found to be leaking (25.553 SCFH) in excess of the Technical Specification 3.7.A.2.b limit of 11.5 SCFH during the performance of Dresden Technical Staff Surveillance Procedure DTS 250-1 "Main Steam Isolation Valve Local Leak Rate Test". An immediate investigation by the cognizant Technical Staff Engineer identified the leakage as coming from the packing on the 2-203-2B valve. An inspection of MSIV 2-203-2B by the Mechanical Maintenance Department found the packing in a slightly degraded condition due to wear experienced from unit operation during the previous fuel cycle. Therefore, on 12-19-86 the valve was repacked and on 12-21-86 an "as left" local leak rate test was successfully performed. The safety significance of this event was minimal because the calculated leakage remained significantly less than the Technical Specification limit for Type B and C leak rate testing which is 493.116 SCFH. Furthermore leakage from the in-line inboard MSIV 2-203-1B remained within Technical Specification limits indicating that the calculated "through" leakage using the "Minimum Pathway" methodology was minimal. The previous failure of an MSIV during an Appendix J local leak rate test is documented in Reportable Occurrence #83-05 on Docket #050237.

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		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Dresden Nuclear Power Station, Unit 2	0 5 0 0 0 2 3 7	8 6	— 0 3 0	— 0 0	0 2 OF 0 2

TEXT *(If more space is required, use additional NRC Form 366A's)* (17)

On December 14, 1986 with Unit 2 at 0% power and in a refueling outage, main steam isolation valve (MSIV, EIIS Code SB) 2-203-2B was found to be leaking in excess of the Technical Specification 3.7.A.2.b limit of 11.5 SCFH during the performance of Dresden Technical Staff Surveillance Procedure (DTS) 250-1 "Main Steam Isolation Valve Local Leak Rate (Dry) Test". The calculated leak rate was determined to be 25.553 SCFH during the test. An immediate investigation by the cognizant Technical Staff Engineer identified the leakage as coming from the valve stem area indicating a packing leak on the 2-203-2B valve. The remaining seven MSIVs successfully completed the Appendix J local leak rate test. Work Request D60399 was initiated for the Mechanical Maintenance Department to inspect MSIV 2-203-2B and adjust the packing as necessary to correct the problem causing the leak.

An inspection of MSIV 2-203-2B (Crane Co. #DR34289-20" Y Pattern Globe Valve) by the Mechanical Maintenance Department found the packing (QP Packing Co.) in a slightly degraded condition due to wear experienced from unit operation during the previous fuel cycle. Therefore, on 12-19-86 the valve was repacked and on 12-21-86 an "as left" local leak rate test (LLRT) was successfully performed. The calculated "as left" leak rate was determined to be 6.578 SCFH. The safety significance of this event was minimal because the calculated leakage remained significantly less than the Technical Specification limit for Type B and C leak rate testing which is 493.116 SCFH. Furthermore leakage from the in-line inboard MSIV 2-203-1B (3.289 SCFH) remained within Technical Specification limits indicating that the calculated "through" leakage using the "Minimum Pathway" methodology was minimal. The previous failure of an MSIV during an Appendix J local leak rate test is documented in Reportable Occurrence #83-05 on Docket #050237.

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EDE LTR #87-009

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Licensee Event Report #86-030-0, Docket #050237 is being submitted as required by Technical Specification 6.6, NUREG 1022 and 10 CFR 50.73(a)(2)(i)(B).

R.A. Flessner for
E.D. Eenigenburg
Station Manager
Dresden Nuclear Power Station

EDE/jmt

Enclosure

cc: J.G. Keppler, Regional Administrator, Region III
File/NRC
File/Numerical

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