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)
La Crosse Boiling Water Reactor (LACBWR)	0 5 0 0 0 4 0 9	1 OF 0 2

TITLE (4)
Type C Leakage Test Failure - Reactor Building Main Steam Isolation Valve and it Bypass

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)
0 1	0 6	8 6	8 6	— 0 0 1	— 0 0	0 1	2 7	8 6	None	0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9)	4	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)			
POWER LEVEL (10)	0 0 0	20.402(b)	20.405(c)	50.73(a)(2)(iv)	73.71(b)
		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 in Abstract below and in Text, NRC Form 366A)
		20.405(a)(1)(iii)	50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	
		20.405(a)(1)(iv)	X 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
Lynne S. Goodman, LACBWR Operations Engineer	AREA CODE 6 0 8 — 6 8 9 - 2 3 3 1

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 V	B 2 9 5	N							
X	S B	I S V	A 1 8 0	N							

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)

During an outage, a Type C leakage test of the Reactor Building Main Steam Isolation Valve and its bypass was conducted. Leakage was measured above the acceptance criteria. The bypass valve was disassembled and a new seat and inner valve installed. It appeared that the valve plug may have been hitting on one side of the seat more than the other. Seating adjustments on the Main Steam Isolation Valve were also made. A test with acceptable results was achieved following the maintenance.

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FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
La Crosse Boiling Water Reactor	0 5 0 0 0 4 0 9	8 6	— 0 0 1	— 0 0	0 2 OF 0 2

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

During a maintenance outage in January 1986, a Type C test was performed on the Reactor Building Main Steam Isolation Valve (SB)(ISV) and its bypass. The test was scheduled to be performed in accordance with the Inservice Testing Program, since the leakage measured during the spring 1985 refueling outage had increased from the 4.8 SCFH measured during November 1983 to 13.5 SCFH, which reduced the margin between the measured leakage rate and maximum permissable rate by more than 50%.

The leakage measured during the test performed January 6, 1986 varied from approximately 40 SCFH to greater than 48 SCFH, which is the top end of the rotameter scale. Per Technical Specification Section 5.2.1.2.c, the combined leakage for all Type B and C tests shall be less than 60% of the Type A allowable test leakage rate. This requirement corresponds to less than 30 SCFH. The combined leakage rate of all other Type B and C tests was approximately 4.65 SCFH.

The Turbine Building Main Steam Isolation Valve was tested. No leakage was measured. It was tagged closed to provide containment integrity for maintenance which was to be performed, which required containment integrity. There is only 1 steam line penetrating containment at LACBWR.

The Reactor Building Main Steam Isolation Valve (RBMSIV) and its 1-1/2 inch bypass are located in parallel sections of the main steam line. They cannot be isolated from each other. The RBMSIV is a 10-inch A-C York, Model No. 5200-BB5-1 Rotovalve. The bypass valve is a BS&B (Black, Sivalls and Bryson), Model No. 70-18-9 air-to-open globe valve. The bypass valve was disassembled. It appeared that the valve plug may have been nesting on one side of the seat more than the other. A new seat, inner valve, seat gasket, pin, stem, bonnet gasket and packing were installed. The Reactor Building Main Steam Isolation Valve was cycled and greased. Its seating adjustments were adjusted. The purpose of the seating adjustments is to adjust the valve seating to compensate for normal wear.

A test with acceptable results was conducted on January 9, 1986. The measured leakage through the RBMSIV and its bypass was 14 SCFH. The cumulative Type B and C tests leakage rate was 18.65 SCFH, well within the acceptance criteria of less than 30 SCFH.

The consequences of this event would not have been excessive even if a Loss of Coolant Accident had occurred while the leakage was above the acceptance criteria. The combined leakage measured during all Type B and C tests was approximately the design basis leakage, 50 SCFH, including the RBMSIV and its bypass. Also, the Turbine Building Main Steam Isolation Valve, which is a remote manual motor-operated valve, would have been available to terminate the leakage through the main steam line valves.

DAIRYLAND Power COOPERATIVE · P.O. BOX 817 · 2615 EAST AV SOUTH · LA CROSSE, WISCONSIN 54601
(608) 788-4000

January 27, 1986

In reply, please
refer to LAC-11384

DOCKET NO. 50-409

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

SUBJECT: DAIRYLAND POWER COOPERATIVE
LA CROSSE BOILING WATER REACTOR (LACBWR)
PROVISIONAL OPERATING LICENSE NO. DPR-45
LICENSEE EVENT REPORT NO. 86-01

Reference: 10 CFR 50.73

Gentlemen:

In accordance with 10 CFR 50.73, attached is Licensee Event Report No. 86-01.

If there are any questions, please contact us.

Sincerely,

DAIRYLAND POWER COOPERATIVE

James W. Taylor
General Manager

JWT:LSG:sks

Attachment

cc: J. G. Keppler, NRC Region III
NRC Resident Inspector
D. Sherman, ANI Library
John Stang, LACBWR Project Manager
INPO

WP6.20.1

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