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)
Cooper Nuclear Station	0 5 0 0 0 2 9 8	1 OF 0 4

TITLE (4)

Excessive Primary Containment Local Leakage Rate

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 5 0 0 0
0 8	1 8	8 5	8 5	— 0 0 5	— 0 0	0 9	1 7	8 5		0 5 0 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)

OPERATING MODE (9)	1				
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)	[X] 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)	[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	
	AREA CODE	
E. M. Mace, Plant Engineering Supervisor	4 0 2	8 2 5 - 3 8 1 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS
B	S B	I S V	A 3 9 5	Y	B	B N	I S V	C 6 3 1	Y
B	B N	I S V	A 5 8 5	Y	B	S J	I S V	A 3 9 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)

Primary Containment local leak rate testing performed during the 1984-85 outage at Cooper Nuclear Station yielded nine type C (isolation valve) and two type B (double o-ring seal) penetration failures, constituting a total "as found" leakage rate in excess of Technical Specification limits. Five isolation valves were replaced and twelve were repaired. Seals were replaced on both type B penetrations and the total leakage was reduced to within specified limits. A program to trend "as found" leakage and make the necessary repairs or replacements has been implemented as a long range plan to reduce "as found" leakage to less than 0.60 La (189 scfh).

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

CNS Surveillance Procedure 6.3.1.1 (Local Leak Rate Testing) was performed during the 1984-85 refueling/IGSCC piping replacement outage in accordance with CNS Technical Specifications Section 4.7.A.2.f and Tables 3.7.2, 3.7.3, and 3.7.4. All Primary Containment isolation valves were closed by their normal means (air or motor operators). Forty-eight type B penetrations and sixty-one type C penetrations were tested. The total "as found" leakage was 720.58 cfh. This exceeds the limit of 189 scfh (0.60 La). Testing was completed on August 18, 1985 with the following penetrations requiring repairs to bring the leakage rate to within established limits.

X-2 Drywell Airlock

Initial leakage was found to be 11.51 cfh. The recommended allowable limit is 6.3 scfh. New gaskets were installed on each door and the airlock was retested, resulting in a reduction of leakage to 3.51 cfh.

X-6 CRD Removal Hatch

Initial testing indicated a leakage rate of 0.49 cfh, which was over the limit of 0.1 scfh. After replacing the o-ring seals, the leakage was reduced to zero.

X-8 Main Steam Line Drain

The test boundary consists of two 3" Anchor gate valves, MS-MO74 and MS-MO77. Initial leakage was 228.57 cfh. Trending shows these valves to have failed LLRT frequently, so both valves were replaced. Subsequent testing resulted in zero leakage.

X-9A RCIC/RWCU Connection to Feedwater

Initial leakage was 219.74 cfh. The seat and disc were lapped and the disc shimmed to improve alignment on RCIC-AO22, an Atwood and Morrill testable check. The soft seat was replaced on RF-15CV, a 24" Anchor tilting disc check. RCIC-MO17, a 1" Conval globe, and RWCU-15CV, a 3" Anchor swing check, were both replaced. Leakage was reduced to 4.83 cfh. The recommended allowable limit has been set at 7.0 scfh.

X-10 RCIC Steam Supply

An initial leakage rate of 6.17 cfh exceeded the limit of 2.0 scfh. The seat and disc were lapped on RCIC-MO15. RCIC-MO16 was replaced. Leakage was reduced to zero. Both valves are 3" Anchor gates.

X-26 Drywell Purge and Vent Exhaust

The established limit was exceeded when initial leakage was determined to be 100.27 cfh. The seat rings were replaced on PC-231MV and PC-246AV. The retest indicated a leakage rate of 5.93 cfh, which is under the 7.5 cfh limit established by ASME Section XI, Article IWV-3426. PC-231MV and PC-246AV are 24" butterfly valves manufactured by Allis-Chalmers.

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

X-39B Drywell Dilution Supply System B

Leakage was determined to be 2.48 cfh initially. ACAD-1311MV, a 1" Anchor gate valve, was disassembled and the gate was lapped. This reduced the leakage rate to 0.26, which conforms to the ASME Section XI, Article IWV-3426 allowable leakage rate of 0.3125 scfh.

X-210B RHR Loop B Minimum Flow

Initial testing indicated a leakage rate of 55.40 cfh. To reduce the rate to below the limit of 1.0 scfh, RHR-MO16B gate was lapped and the seats cleaned. Retesting showed a leakage rate of 0.43 cfh. RHR-MO16B is a 4" Anchor gate valve.

X-214 HPCI Turbine Exhaust

Initial leakage was 8.67 cfh, compared to the established limit of 3.0 scfh. HPCI-15CV was disassembled, the seats lapped and the disc machined. The stellite seating surface was refurbished on HPCI-44 and the results of the retest were 1.44 cfh. HPCI-15CV is a 20" Anchor swing check. HPCI-44 is a 20" Anchor stop check.

X-214 Suppression Chamber Purge and Vent Exhaust

Seat rings were replaced on PC-230MV and PC-245AV to bring the initial leakage rate of 35.91 cfh to below the limit of 5.0 scfh. Actual leakage after repairing the 24" Allis-Chalmers butterfly valves was 2.28 cfh.

X-223B Core Spray Pump B Minimum Flow

Initial leakage was 3.53 cfh. New seats were installed and the gate was lapped on CS-MO5B, bringing the leakage rate to 0.07 cfh. The limit is 1.0 scfh. CS-MO5B is a 3" Anchor gate.

The total "as left" leakage rate after repairs were made is 67.15 cfh. Although the 1984-85 "as found" leakage of 720.58 cfh was over the limit of 189 scfh, it was reduced by a factor of six from the 1983 total. UE 75-22 and LERs 77-57, 79-9, 80-12, 81-15, 82-15, and 83-13 describe previous findings of this nature. There were no significant occurrences or adverse effects to public health or safety as a result of this event.

Plans are to continue in an effort to reduce the "as found" leakage to below 189 scfh through preventive maintenance and trending to determine isolation valves in need of attention.

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LICENSEE EVENT REPORT (LER) FAILURE CONTINUATION

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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	
B	JM	ISV	A395	Y							
B	BN	ISV	A395	Y							
B	BB	ISV	A180	Y							
B	BB	ISV	A395	Y							
B	BO	ISV	A395	Y							
B	BJ	ISV	A395	Y							
B	BM	ISV	A395	Y							
B	NH	PENT	C310	Y							

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Nebraska Public Power District

COOPER NUCLEAR STATION
P.O. BOX 98, BROWNVILLE, NEBRASKA 68321
TELEPHONE (402) 825-3811

CNSS850521

September 17, 1985

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

Dear Sir:

Cooper Nuclear Station Licensee Event Report 85-005 is forwarded as an attachment to this letter.

Sincerely,

P. V. Thomason
Division Manager of
Nuclear Operations

PVT:lb
Attach.
cc: R. D. Martin
L. G. Kuncl
J. D. Weaver
L. R. Berry
INPO Records Center
ANI Library

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