

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1) Cooper Nuclear Station	DOCKET NUMBER (2) 0 5 0 0 0 2 9 8	PAGE (3) 1 OF 0 5
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TITLE (4)
Excessive Primary Containment Leakage
Discovered During Local Leak Rate Testing

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		
0	1	0	8	8	7	8	7	8	-		
			0	0	4	0	0	2	0 6 8 7		
									DOCKET NUMBER(S) 0 5 0 0 0		

OPERATING MODE (9) N	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)(vii)(B)					
20.405(a)(1)(v)			50.73(a)(2)(iii)			50.73(a)(2)(ix)						

LICENSEE CONTACT FOR THIS LER (12)									
NAME D. L. Reeves, Jr.							TELEPHONE NUMBER		
							AREA CODE 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	B J	I S V	H 1 9 5	Y	B	B J	I S V	A 3 9 5	Y	
B	N H	P E N	C 3 1 0	N	B	J M	I S V	A 3 9 5	Y	

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

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single space typewritten lines) (16)

Performance of local leak rate testing accomplished during the 1986 refueling/major maintenance outage resulted in a determination of total "as-found" leakage of 3314.09 scfh. The total allowable leak rate as specified in the CNS Technical Specifications, paragraph 4.7.A.2, is 189 scfh. The major contributors to the excessive leak rate were found to be the Main Feedwater check valves, the Reactor Water Cleanup System (RWCU) Return to Reactor Vessel check valve and the High Pressure Coolant Injection (HPCI) Pump Suction from Suppression Pool isolation valve. An additional eighteen valves were also determined to be leaking in excess of their individual allowable rates.

Corrective maintenance was performed on all of the valves determined to be deficient. Retesting confirmed the adequacy of all repairs made. The total calculated local leak rate, based upon leak rate testing accomplished following completion of repairs, was determined to be 63.29 scfh.

Specific corrective action which is planned includes evaluation of the adequacy of the seat ring material in the Feedwater check valves and an increase in the seat ring replacement frequency from three years to each refueling outage. The two remaining valves which were identified as major contributors will be further evaluated through the Local Leak Rate Trend Program and the Nonconformance Reporting Program.

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TEXT (If more space is required, use additional NRC Form 388A's) (17)

A. EVENT DESCRIPTION

Performance of local leak rate testing accomplished during the 1986 refueling/major maintenance outage resulted in a determination of total "as-found" leakage of 3314.09 scfh. The total allowable leak rate, as specified in CNS Technical Specifications, paragraph 4.7.A.2, is 189 scfh. The major contributors to the excessive leak rate were found to be the Main Feedwater check valves (RF-CV-16CV, 15CV, 14CV, and 13CV), the Reactor Water Cleanup System (RWCU) Return to Reactor Vessel check valve (RWCU-CV-15CV), and the HPCI Pump Suction from Suppression Pool isolation valve (HPCI-MOV-58MV). In accordance with Procedure 6.3.1.1, Local Leak Rate Testing, the allowable combined leakage rate through these valves is 35 scfh. The as-found leakage was determined to be 3094.60 scfh, accounting for 93% of the total as-found leakage. In addition to these six identified deficiencies, a total of eighteen additional primary containment valves were determined to be leaking in excess of their individual allowable rates, contributing 177.98 scfh to the total leakage. The combined allowable leakage for these valves is 29.60 scfh.

B. PLANT STATUS

Shutdown for a refueling/major maintenance outage which had commenced on October 4, 1986. Prior to performance of the local leak rate testing on those isolation valves which serve as the primary boundary, the valves were closed using their normal means of cycling.

C. BASIS FOR REPORT

Excessive primary containment leakage; hence, reportable in accordance with 10CFR50.73, paragraph (a)(2)(ii).

D. CAUSE OF EVENT

The cause attributed to the Main Feedwater check valve excessive leak rate is believed to be due to the overestimated lifetime of the seat ring material. In 1983, in an effort to resolve past local leak rate testing concerns, a soft seat material was installed in these valves. At the time, a preventive maintenance schedule was established to replace this material every third outage. During this most recent test, the leak rates for each of the four Feedwater check valves were unsatisfactory. Hence, the purported lifetime of the seat ring material was not realized and a replacement interval of three years is now considered to be unsatisfactory.

The Reactor Water Cleanup check valve (RWCU-CV-15CV) was replaced in 1984. Following replacement, an acceptable leak rate test was performed. However, the valve was found to be deficient during this most recent round of testing. During inspection, a slight deformity in the flapper seat was believed to exist. The flapper seat was lapped and, following reassembly, an acceptable leak rate test was performed.

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The cause of the excess leakage through HPCI-MOV-58MV could not be precisely determined. Maintenance had been performed on the valve in 1983 when the seats were re-stellited and the gate polished. The valve successfully passed the leak rate test conducted in 1984. Following the unsuccessful initial test during the 1986 outage, the valve was disassembled and the seats and gate were lapped. The valve was reassembled and successfully passed the leak rate test. No specific deficiencies or observations regarding the condition of the seats or gate were noted.

Finally, with respect to the eighteen other valves which were determined to be leaking in excess of the allowable limits, the cause of the majority of the failures is believed to be due to wearing of seating surfaces which might reasonably be expected.

E. SAFETY SIGNIFICANCE

Numerically, the as-found leakage determined from the local leak rate testing constitutes a leak rate in excess of ten times the allowable test leak rate defined in the Bases for Technical Specifications, paragraphs 3.7.A.2 and 4.7.A.2 (page 177). Consequently, under LOCA conditions, the potential for release of fission products to attached fluid systems or directly to Secondary Containment would be markedly increased.

In Chapter XIV of the USAR, paragraph 6.3.6, Fission Product Release to Environs, it is assumed that in calculating the release to the environs, the design leakage is directly from Primary to Secondary Containment. With respect to the discrepancies (failures) determined during local leak rate testing, a substantial majority of the excessive leak rate was not associated with Primary to Secondary penetrations, but rather constituted Reactor Coolant System pressure boundary (isolation valve) leakage to attached systems; e.g., Main Feedwater, HPCI, and RWCU. Only a minor portion of the leakage constituted direct leakage from Primary to Secondary Containment. Hence, on a more practical basis, the safety significance of these leak rate testing failures is considered to be substantially less than that which is inferred from a purely numerical perspective.

F. CORRECTIVE ACTION

All of the discrepancies identified during the performance of the local leak rate testing were corrected and satisfactorily retested. With respect to the six most significant testing failures, new seat rings were installed in the four Feedwater Check Valves and, as previously noted, the seating surfaces for the RWCU and HPCI valves were lapped. Maintenance action performed on the eighteen other penetrations, following which successful leak rate tests were accomplished, primarily involved replacement of seat rings or lapping of seating surfaces. As a result of these maintenance actions, the leak rate attributed to the testing discrepancies was reduced from 3272.58 scfh to 49.68 scfh. The total calculated local leak rate following repairs was calculated to be 63.29 scfh.

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TEXT (If more space is required, use additional NRC Form 368A's) (17)

Specific corrective action which is planned includes evaluation of the adequacy of the seat ring material in the Feedwater check valves and an increase in the seat ring replacement frequency from three years to each refueling outage. The two remaining valves which were identified as major contributors will be further evaluated through the Local Leak Rate Trend Program and the Nonconformance Reporting Program.

G. PAST SIMILAR EVENTS

LER 85-005, Revision 1, dated January 16, 1986.



Nebraska Public Power District

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

CNSS870064

February 6, 1987

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

Dear Sir:

Cooper Nuclear Station Licensee Event Report 87-004 is forwarded as an attachment to this letter.

Sincerely,

A handwritten signature in cursive script, appearing to read "G. R. Horn".

G. R. Horn
Division Manager of
Nuclear Operations

GRH:lb

Attach.

cc: R. D. Martin
L. G. Kunc1
K. C. Walden
C. M. Kuta
INPO Records Center
ANI Library

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