Nebraska Public Power District

COOPER NUCLEAR STATION
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CNSS933077

April 9, 1993

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

Dear Sir:

Cooper Nuclear Station Licensee Event Report 93-005, Revision 0, is being forwarded as an attachment to this letter.

Sincerely,

R. L. Gardner
Plant Manager

RLG/ju

Attachments

cc: J. L. Milhoan
G. R. Horn
J. M. Meacham
R. E. Wilbur
V. L. Wolstenholm
D. A. Whitman
INPO Records Center
NRC Resident Inspector
R. J. Singer
CNS Training
CNS Quality Assurance

JE22 1/1

NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/88

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 Leakage Discovered During Local Leak Rate Testing of Reactor Feedwater Check Valves

EVENT DATE (5) MONTH	DAY	YEAR	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	REPORT DATE (7) MONTH	DAY	YEAR	OTHER FACILITIES INVOLVED (8) FACILITY NAMES	DOCKET NUMBER(S)
0 3	1 0	9 3	9 3	0 0 5	0 0	0 4	0 9	9 3		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): N

POWER LEVEL (10): 0 0 0

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

[ ] 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)	[X] 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 AREA CODE	
John R. Myers	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	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
B	S J	I S V	A 3 9 5	Y					

SUPPLEMENTAL REPORT EXPECTED (14)

[ ] YES (If yes, complete EXPECTED SUBMISSION DATE) [X] NO

EXPECTED SUBMISSION DATE (15): MONTH | DAY | YEAR

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

On March 10, 1993, upon completion of the Local Leak Rate Test (LLRT) of the Reactor Feedwater check valves, the Primary Containment leak rate was determined to be in excess of 1.0 La, where La is the allowable containment leak rate. The leakage of the Reactor Feedwater check valves was too great to be accurately quantified using the available test equipment. This testing was conducted with the plant in Cold Shutdown.

The Reactor Feedwater check valve leakage was attributable to degraded resilient seat material, body-disc misalignment, and improper clearances of the valve internals. Additionally, draining of the piping system is restricted by the available vents and drains, potentially affecting the as-found seating of the valves. Modification and repair efforts include line-boring the valve body and disc, installing new hinge pins and bushings with tighter clearances, replacing the seating materials, and adding additional vents and drains to the system. The valve manufacturer is assisting in the evaluation, modification, and repair of the valves. These efforts were based on a review of past valve performance and investigation of the performance of similar valves at other plants.

LLRTs will be completed with satisfactory results prior to startup from the 1993 Refueling Outage.

A. Event Description:

On March 10, 1993, upon completion of the Local Leak Rate Test (LLRT) of the Reactor Feedwater check valves, the as-found Primary Containment leak rate was determined to exceed 1.0 La, where La is the allowable containment leak rate. The leakage of the Reactor Feedwater check valves was too great to be accurately quantified using the available test equipment. A review of the as-found leakage rate of other components tested during this outage indicates that, had the Reactor Feedwater check valves performed acceptably, the total as-found leakage would not have exceeded the allowable containment leak rate.

B. Plant Status

Determination that the LLRT results exceeded 0.60 La, the test limit specified in paragraph 4.7.A.2.f.1 of the Technical Specifications, occurred immediately following completion of the LLRT on March 10, 1993. At this time, the plant was in cold shutdown. It was also recognized that the total as-found Primary Containment leak rate exceeded 1.0 La, and NRC notification was made in accordance with 10CFR50.72.

C. Basis for Report

A condition found while shutdown wherein a principal safety barrier (Primary Containment) was determined to be seriously degraded, reportable in accordance with 10CFR50.73(a)(2)(ii). In addition, the LLRT results are being reported in accordance with 10CFR50.73(a)(2)(i)(B), a condition prohibited by Technical Specifications.

D. Cause

Equipment Deficiency. When originally installed, these valves were intended to be tested with water as the test media. In 1983, as a result of the issuance of Appendix J to 10CFR50, the test media was changed to air. Upon testing with air, problems were immediately experienced meeting the acceptance criteria. Although the valves have been shown to be capable of providing acceptable test results following refurbishment, these results demonstrate that the valves are subject to long term degradation which increases the likelihood of test failures. Additionally, minor alignment changes of the valve as a result of normal wear can result in LLRT performance problems. An evaluation of the present situation indicates that performance has gradually degraded.

During maintenance following the failed LLRT, three resilient seat rings were found to be significantly degraded, with the fourth only slightly degraded. Indications are that the discs may have been out of alignment. Additionally, the internal parts for the valve have been supplied with substantially greater clearances than the repair parts currently being installed, also creating the potential for alignment problems. The manufacturer's representative noted that the resilient seat may have protruded too far from the disc.

E. Safety Significance

The greatest release of radioactive material to the Primary Containment results from a complete circumferential break of one of the recirculation loop lines, the Design Basis Accident Loss of Coolant Accident (DBA-LOCA). The potential off-site radiological dose is conservatively based on a Primary Containment leakage of La (316 SCFH, which is equal to 0.635 percent by weight per day of the containment volume at 58 psig) for 30 days. Consequently, under DBA-LOCA conditions the potential for release of fission products would be markedly increased.

F. Safety Implications

The safety implications of this event are fully described in the section above.

G. Corrective Action

To provide better alignment between the valve body and disc, the valves and discs are being precision line-bored. Additionally, new hinge pins and bushings with closer clearances are being installed. To ensure the resilient seat is not interfering with the seating of the disc, the groove depth is being increased such that the protrusion of the resilient seat is less, and new resilient seats are being installed. Additional vents and drains for the system are also being installed to enhance testing. The valve manufacturer is assisting in the evaluation, modification, and repair of the valves. These efforts were based on a review of past valve performance and investigation of the performance of similar valves at other plants. LLRTs will be completed with satisfactory results prior to startup.

As a followup to LER 91-020, a detailed evaluation of the failure history, past maintenance practices, potential causes, and actions to resolve the identified problems meeting leak rate requirements was undertaken. The results of this evaluation indicated that a few penetrations had experienced recurring problems meeting LLRT requirements. The Reactor Feedwater check valves were among the components identified. Based on the success of other utilities, it was determined that better alignment of the valves could result in successful test performance. Therefore, it was decided to perform the alignment during the 1993 refueling Outage upon failure to meet the LLRT requirements. The addition of vents and drains had been scheduled for the 1994 outage, but was expedited to accompany the work in progress.

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

H. Similar Events

Since 1984, three LERs associated with LLRT have been submitted:

LER	DATE	TITLE
85-005, Rev. 1	01/16/86	Excessive Primary Containment Local Leakage Rate
87-004, Rev. 0	02/06/87	Excessive Primary Containment Leakage Discovered During Local Leak Rate Testing
91-020, Rev. 0	01/09/92	Failure of the Primary Containment Integrated Leak Rate Test due to Drywell Vent Monitor System and Containment Penetration Leakage

Of these LERs, the Reactor Feedwater check valves are noted as contributing to the as found leakage in LERs 87-004 and 91-020.

Supplemental Information

The Reactor Feedwater Check Valves are 18 inch diameter tilting disc check valves manufactured by Anchor Darling Valve Co., Model 920-3.