Nebraska Public Power District

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NLS950064

March 3, 1995

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

Dear Sir:

Cooper Nuclear Station Licensee Event Report 95-004 is forwarded as an attachment to this letter.

Sincerely,

J. T. Herron
Plant Manager

/nr

Attachment

cc: L. J. Callan
G. R. Horn
J. H. Mueller
R. G. Jones
R. A. Sessoms
K. C. Walden
R. L. Koch
INPO Records Center
NRC Resident Inspector
R. J. Singer
CNS Training
CNS Quality Assurance

IE22

NRC FORM 366 (5-92) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 5/31/95

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
COOPER NUCLEAR STATION	05000298	1 OF 3

TITLE (4) Primary Containment Group Isolations Caused by Surveillance Procedure Deficiencies

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 NAME	DOCKET NUMBER
02	01	95	95	-- 004 --	00	03	03	95	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	N	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)							
			20.402(b)		20.405(c)	X	50.73(a)(2)(iv)		73.71(b)
POWER LEVEL (10)	000		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
			20.405(a)(1)(iii)		50.73(a)(2)(i)		50.73(a)(2)(viii)(A)		(Specify in Abstract below and in Text, NRC Form 366A)
			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 (Include Area Code)
R. A. Dieterich, Licensing Engineer	(402) 825-3811

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

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 15 single-spaced typewritten lines) (16)

On February 1, 1995, at approximately 8:30 am, Primary Containment Groups 2,3 and 6 isolations and a Standby Gas Treatment System initiation occurred during the performance of Surveillance Procedure (SP) 6.3.10.2, "Instrument Line Excess Flow Check Valve Test." A reactor hydrostatic test was in progress with reactor pressure being maintained at 1025 psig with the Control Rod Drive Pumps and Reactor Water Cleanup (RWCU) System. The isolation signal isolated the RWCU system and reactor pressure increased to 1060 psig before pressure control was restored by use of steam line drains. All actuations and isolations occurred as required, the reactor vessel pressure limit was not reached, and no relief valves opened.

Instructions in the procedure erroneously indicated that sections 8.1 through 8.7 could be performed in any order or simultaneously. In the past, they were successfully performed consecutively or simultaneously. However, on this occasion, section 8.7 was performed before section 8.6 and as a result, there was no isolation of the instrumentation that created the pressure perturbation and initiated the group isolations. Further investigation revealed that section 8.7 specified removal of the incorrect instrument from service and as a result, check valve NBI-CV-19BCV had not been tested since 1977.

A change to the procedure in 1977 was inadequately reviewed and implemented an incorrect valve configuration. Since that time, a more rigorous review process has been initiated including walkdowns to verify proper valve lineups.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
COOPER NUCLEAR STATION	05000298	95	-- 004 --	00	2 OF 3

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

Plant Status

The plant was shutdown with a reactor hydrostatic test in progress. Reactor pressure was at 1025 psig and was being maintained with the Control Rod Drive (CRD) pumps and Reactor Water Cleanup (RWCU) system.

Event Description

On February 1, 1995, at approximately 8:30 am, Primary Containment Group 2 (Containment Vent Valves, TIP Valves and Residual Heat Removal Shutdown Cooling Mode), Group 3 (Reactor Water Cleanup), Group 6 (Reactor Building Ventilation) isolations, and a Standby Gas Treatment System initiation occurred during the performance of Surveillance Procedure (SP) 6.3.10.2, "Instrument Line Excess Flow Check Valve Test." A reactor hydrostatic test was in progress in accordance with SP 6.3.10.28, "ASME Class 1-N System Leakage Test." Reactor pressure was being maintained at 1025 psig with the CRD pumps and RWCU system.

The isolation signal isolated the RWCU System and reactor pressure increased to 1060 psig before pressure control was restored by use of steam line drains. The excess flow check valve procedure was terminated at this point and reviewed for deficiencies. It was determined that the procedure allowed for completion of sections out of sequence; however, the procedural valve lineups did not support this. Instructions in the procedure indicated that sections 8.1 through 8.7 could be performed in any order or simultaneously. In the past sections 8.1 through 8.7 were performed simultaneously or consecutively. The instrumentation that created the pressure perturbation and initiated the group isolations would have been isolated by section 8.6 of the procedure had it been performed first. On this occasion, section 8.7 was performed first since this group was the most likely to require maintenance. The test was performed with Reactor Low Water Level Instrument NBI-LIS-83A still in service. When the drain valve was closed it caused a perturbation of the water level reference leg which caused the group isolations.

Further investigation revealed that section 8.7 specified removal of the incorrect instrument from service. This deficiency resulted in testing excess flow check valve NBI-CV-15BCV twice and prevented testing excess flow check valve NBI-CV-19BCV. It was subsequently determined that check valve NBI-CV-19BCV had not been tested since 1977 because of this deficiency.

All actuations and isolations occurred as required. The reactor vessel pressure limit was not reached and since Safety Relief Valve (SRV) setpoints were not reached, no SRVs opened. All isolations and actuations were reset by approximately 8:41 am. Subsequently, Excess Flow Check Valve NBI-CV-19BCV was tested satisfactorily.

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

Safety Significance

The safety significance of this event is minimal. The group isolations that caused the pressure increase also served to alert the plant operators of the problem which provided time to re-establish pressure control prior to exceeding any safety limits or relief valve setpoints. This test is always performed while the reactor is shutdown, so the margin of safety was not compromised. The failure to test excess flow check valve NBI-CV-19BCV since 1977 also had minimal impact on safety since the "As-Found" leakage was within required limits.

Cause

A modification to the Nuclear Boiler Instrumentation System in 1977 resulted in changes to SP 6.3.10.2. These changes implemented an incorrect valve configuration for testing excess flow check valve NBI-CV-19BCV. At that time the procedure change program was inadequate with regard to review of procedure changes. Since that time, performances of SP 6.3.10.2 did not cause group isolations due to the manner in which the tests were performed. Instructions in the procedure indicated that sections 8.1 through 8.7 could be performed in any order or simultaneously. In the past, sections 8.1 through 8.7 were performed simultaneously or consecutively. The apparent successful performance of the test for many years resulted in a mindset that the test was correct. On this occasion, section 8.7 was performed first since this group was the most likely to require maintenance, and the test was performed with instrumentation still in service.

Corrective Action

SP 6.3.10.2 was terminated and the correct valve lineup was determined. The surveillance procedure was changed was to rectify the discrepancies, and NBI-CV-19BCV was tested satisfactorily.

The Plant Manager has conducted briefings with I&C technicians stressing the importance of a questioning attitude while performing test procedures instead of blind confidence based on past experience. This attitude is also stressed in self-checking training with operations and maintenance personnel.

The procedure change process has evolved since 1977. It now requires a more rigorous review of procedure revisions including reviews by engineering and the establishment of procedure owners with overall responsibility for revisions. Procedure changes are now evaluated to determine if a walkdown is required as part of the review and approval process. Procedure changes resulting from changes to permanent component lineups require such a walkdown.

LIST OF NRC COMMITMENTS	ATTACHMENT 3

Correspondence No: NLS950064

The following table identifies those actions committed to by the District in this document. Any other actions discussed in the submittal represent intended or planned actions by the District. They are described to the NRC for the NRC's information and are not regulatory commitments. Please notify the Licensing Manager at Cooper Nuclear Station of any questions regarding this document or any associated regulatory commitments.

COMMITMENT	COMMITTED DATE OR OUTAGE
None	