Nebraska Public Power District

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

CNSS948116

April 15, 1994

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

Dear Sir:

Cooper Nuclear Station Licensee Event Report 94-005 is forwarded as an attachment to this letter.

Sincerely,

Jerry Gaper for
R. L. Gardner
Plant Manager

RLG/nc

Attachment

cc: L. J. Callan
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

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 5

TITLE (4) Actuation of Shutdown Cooling Isolation Valves Due to a Pressure Perturbation Caused By Void Collapse in the RHR System

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
										05000
03	17	94	94	-- 005 --	00	04	15	94	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	N
POWER LEVEL (10)	0

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)
	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)
John R. Myers	(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 March 17, 1994, at 9:26 a.m., with the Reactor at approximately atmospheric pressure and a temperature of 175 degrees Fahrenheit in the Reactor Recirculation system, the Shutdown Cooling Supply Isolation Valves (Group 2 components) automatically closed. The Residual Heat Removal system was in operation in the Shutdown Cooling mode at the start of a planned maintenance outage. Both Shutdown Cooling isolation valves closed as a result of the spurious actuation of the Shutdown Cooling high suction pressure interlock. Shutdown Cooling was restored in 13 minutes and the Reactor cooled to 110 degrees without further difficulties.

The most probable cause of this event was steam void formation and collapse in the RHR system piping. The NSSS Vendor indicated that void formation is not unexpected at this temperature. Equipment which could potentially have caused this event was inspected or tested to verify an equipment problem was not the cause. Piping and equipment were inspected to verify no damage had occurred. Procedures and practices will be reviewed, including a review of the practices of other similar vintage BWRs, and enhancements implemented as appropriate.

A. Event Description

On March 16, at 7:45 p.m., the plant was removed from service to perform planned maintenance on the A loop of the Residual Heat Removal (RHR) system. On March 17, at approximately 6:58 a.m., the B loop of RHR was placed in the Shutdown Cooling mode of operation. After the reactor coolant temperature was reduced below 212 degrees Fahrenheit, the Reactor head vents were opened. A mechanical vacuum pump was operating to maintain condenser vacuum for cooldown, and the A Reactor Recirculation (RR) loop was in operation.

At 9:26 a.m., the Shutdown Cooling Supply Isolation Valves, RHR-MOV-M017 and RHR-MOV-M018, closed, and the D RHR pump, which was in operation, tripped. Abnormal Procedure 2.4.2.4.1, Loss of Shutdown Cooling, was entered, and the automatic actions were verified to have occurred. Just prior to the closure of the valves, operating personnel noted that Reactor vessel level was decreasing. It was subsequently determined that level had dropped approximately 7 inches. After verifying all Reactor parameters were appropriate for the existing conditions and that the valves fully closed, the Group 2 relay was reset, the isolation valves were reopened, and shutdown cooling recommenced at 9:39 a.m. A review by operating and engineering personnel determined that the valve closure resulted from actuation of the RHR pump high suction pressure interlock, associated with pressure switches RR-PS-128A and RR-PS-128B, which actuate at a vessel dome pressure at or below 75 psig to protect the piping between the isolation valves and the RHR pump suction. Although the high suction pressure interlock does not actuate the Group 2 logic (the actuating signals are low Reactor water level or high drywell pressure), the resetting of the circuit associated with the high pressure interlock relay utilizes the Group 2 reset switch. Reactor coolant temperature increased from 175 to 181 degrees Fahrenheit during the period shutdown cooling was not operating. The Reactor was subsequently cooled to approximately 110 degrees.

B. Plant Status

The plant was beginning a planned maintenance outage, with the Reactor at approximately atmospheric pressure and a temperature of 175 degrees Fahrenheit in the Reactor Recirculation system. Reactor vessel level was at approximately 55 inches, 219 inches above the top of active fuel.

C. Basis for Report

Actuation of the Shutdown Cooling isolation valves, part of the Group 2 primary containment isolation. Closure of these valves was due to actuation of the Shutdown Cooling high suction pressure interlock, a non-ESF signal. This event is being reported in accordance with the criteria prescribed by 10CFR50.73(a)(2)(iv).

D. Cause

The most probable cause of this event was steam void formation and collapse in the injection piping of the idle A RHR loop (i.e. water hammer). With water in the piping at or near saturation temperature, low pressures in the Reactor Coolant System allowed void formation in the RHR lines. Operation of the mechanical vacuum pump created a partial vacuum condition, making void formation more likely. Void collapse caused a pressure perturbation which was sensed by the high suction pressure switches. This cause is supported by personnel observations and recorded data. Personnel at the idle RHR loop heard a "bang" and felt pipe vibrations, and instrumentation for the A RR pump recorded increased vibration. A technician in the vicinity of the D RHR pump did not note any unusual noise prior to trip of the pump, indicating that the operating loop did not experience cavitation. Reactor vessel level decreased by approximately 7 inches (to fill the void), and other indications consistent with pressure perturbations were noted.

An independent review of the event by General Electric indicated that void formation is not unexpected at this temperature. This review confirmed that the cause identified was the most probable, and also identified other BWRs of the same vintage which had experienced shutdown cooling isolations under similar conditions.

E. Safety Significance

The increase in Reactor coolant temperature was limited to 6 degrees during the 13 minute period shutdown cooling was not operating.

The Shutdown Cooling high suction pressure interlock provides overpressure protection for the piping between the Shutdown Cooling isolation valves and the RHR pump suction. The actual setpoint of 72.5 psig ensures that the pressure in the piping remains below the design pressure of 150 psig. Upon sensing a momentary pressure perturbation, the pressure switches initiated the closure of the valves as designed. Inspection of the piping during the subsequent maintenance outage found no evidence of pipe motion which would indicate potential damage. The independent review conducted by General Electric indicated the magnitude of the pressure was approximately 296 psig at the point of formation, and 75 psig at the isolation switch location. Confirmation of this is provided by the lack of annunciation of a high pressure condition in the RHR pump suction, as provided by pressure switch RHR-PS-118. This switch is set to actuate at or below 100 psig.

F. Safety Implications

The identified safety function of these valves is to close to protect the suction piping to the RHR pumps from excessive pressure. This function was fulfilled since the valves closed on a momentary high pressure signal. Additional means are available to provide the required decay heat removal capability should a high suction pressure actuation occur, and guidance exists to accommodate such isolations should they occur.

G. Corrective Action

Shutdown cooling was restored as noted above, and the Reactor coolant temperature was reduced to a value at which additional isolations were unlikely to occur prior to the start of maintenance activities. A Problem Resolution Team was formed to investigate the event and recommend actions to prevent recurrence. The team recommendations included the following actions:

1. Review the list of "protected systems" associated with the work scheduled for the A loop of RHR to identify systems that should not be worked on to prevent shutdown cooling isolations and that could provide alternate means of decay heat removal. The initial list had been identified prior to the initiation of the maintenance shutdown. The initial list was verified to be complete.

2. Inspect piping potentially affected by the water hammer. Piping outside the drywell was walked down immediately, and that inside upon completion of de-inerting. No problems were found.

3. Check the calibration of high suction pressure interlock switches RR-PS-128A and B. The switches were found to be within calibration tolerances.

4. Check the tightness of the electrical connections for the associated relays. The connections were found to be tight.

5. Contact other BWR plants of a similar vintage to determine additional actions which may be available to minimize the potential for a similar water hammer, and revise procedures accordingly. The acceptability of limiting the maximum RHR shutdown cooling flow, a means to provide cooling flow for the idle hot RHR injection line, and the acceptability of limiting the operation of the mechanical vacuum pumps after the Reactor has been vented will be investigated.

G. Corrective Action (continued)

6. Revise procedures and guidelines to ensure that work which would potentially affect the idle RHR loop is not commenced until appropriate conditions are reached.

7. Enhance the abnormal procedure for the loss of shutdown cooling to incorporate the lessons learned from this event.

8. The NSSS Vendor will be advised of this event and requested to investigate any revisions to the recommendations of SIL 175.

H. Similar Events

LER 92-007 identifies a shutdown cooling isolation which occurred with reactor coolant at a temperature of approximately 240 degrees and low pressure, while warming the RHR system prior to placing it in service. Corrective actions for that event included procedure revisions to require pressure to be maintained until shutdown cooling was in service, which typically occurs at a Reactor coolant temperature of 212 degrees, to prevent void formation. Based on industry information available at the time, void formation was not anticipated to occur at the temperatures existing during the 1994 event.