

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1) <b>Cooper Nuclear Station</b>	DOCKET NUMBER (2) <b>0 5 0 0 0 2 1 9 8</b>	PAGE (3) <b>1 OF 0 4</b>
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TITLE (4) **Reactor Scram and Group Isolations Due to Low Reactor Vessel Level During Troubleshooting of Apparent Reversed Leads to the Steam and Feed Flow Recorder**

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)		
<b>0 1 0 7 8 7 8 7</b>	<b>-</b>	<b>0 0 3</b>	<b>-</b>	<b>0 0</b>	<b>0 2 0 6 8 7</b>				<b>0 5 0 0 0</b>		
									<b>0 5 0 0 0</b>		

OPERATING MODE (9) <b>N</b>	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR § (Check one or more of the following) (11)									
POWER LEVEL (10) <b>0 3 0</b>	<input type="checkbox"/> 20.402(b)	<input type="checkbox"/> 20.405(c)	<input checked="" type="checkbox"/> 50.73(a)(2)(iv)	<input type="checkbox"/> 73.71(b)						
	<input type="checkbox"/> 20.405(a)(1)(i)	<input type="checkbox"/> 50.36(c)(1)	<input type="checkbox"/> 50.73(a)(2)(v)	<input type="checkbox"/> 73.71(c)						
	<input type="checkbox"/> 20.405(a)(1)(ii)	<input type="checkbox"/> 50.36(c)(2)	<input type="checkbox"/> 50.73(a)(2)(vii)	OTHER (Specify in Abstract below and in Text, NRC Form 366A)						
	<input type="checkbox"/> 20.405(a)(1)(iii)	<input type="checkbox"/> 50.73(a)(2)(i)	<input type="checkbox"/> 50.73(a)(2)(viii)(A)							
	<input type="checkbox"/> 20.405(a)(1)(iv)	<input type="checkbox"/> 50.73(a)(2)(ii)	<input type="checkbox"/> 50.73(a)(2)(viii)(B)							
<input type="checkbox"/> 20.405(a)(1)(v)	<input type="checkbox"/> 50.73(a)(2)(iii)	<input type="checkbox"/> 50.73(a)(2)(x)								

LICENSEE CONTACT FOR THIS LER (12)

NAME <b>D. L. Reeves, Jr.</b>	TELEPHONE NUMBER
	AREA CODE <b>4 0 2</b>
	<b>8 2 5 - 3 8 1 1</b>

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NRPDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NRPDS

SUPPLEMENTAL REPORT EXPECTED (14)

<input type="checkbox"/> YES (If yes, complete EXPECTED SUBMISSION DATE) <input checked="" type="checkbox"/> NO	EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR

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

On January 7, 1987, during power escalation following the 1986 refueling/major maintenance outage, it became apparent that the input leads to the Steam and Feedwater Flow Recorder (RFC-RF-96) may have been reversed during Detailed Control Room Design Review (DCRDR) implementation. While troubleshooting the apparent lead reversal, the flow signal to the recorder and to the Reactor Water Level Control System was lost. The loss of signal caused the operating Reactor Feedwater Pump speed and flow to increase substantially, exceeding the capacity of the operating Condensate and Condensate Booster Pumps. Consequently, the Reactor Feed Pump tripped due to low suction pressure. Upon loss of feedwater, the reactor water level decreased until, at 1945, a Reactor Scram and Group Isolations 2, 3 and 6 occurred. All safety systems operated as designed, control of reactor vessel level was reestablished, and normal scram recovery procedures were followed.

Upon investigation, it was determined that the recorder RFC-RF-96 input lead terminals had not been fully tightened during performance of the aforementioned DCRDR project. Hence, when a voltmeter was connected to verify the apparent lead reversal, the circuit continuity was interrupted and the feed flow signal lost, initiating the feedwater transient. Therefore, the root cause of the event has been determined to be due to an apparent insufficient acceptance test of the Steam and Feedwater Flow Recorder (RFC-RF-96) during and following completion of the DCRDR design change activity. Corrective action taken following the scram and prior to restart included a verification of terminal connections on Panel 9-5 and in the Feedwater and High Pressure Coolant Injection (HPCI) control systems to ensure tightness. No other discrepancies were noted. Further corrective action to be taken includes a review of this event with all licensed operators and a review of the acceptance testing and quality control program for this particular Design Change.

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LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

FACILITY NAME (1)  Cooper Nuclear Station	DOCKET NUMBER (2)  0 5 0 0 0 2 9 8 8 7 -	LER NUMBER (6)			PAGE (3)	
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER		
		0 0 3 -	0 0 0	0 0 0	0 2	OF 0 4

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

A. EVENT DESCRIPTION

On January 7, 1987, while an Instrument and Control Technician was connecting a voltmeter to the feedwater flow leads to the Steam and Feedwater Flow Recorder (RFC-RF-96) on panel 9-5 in the Control Room to investigate an apparent reversal of leads to this recorder, one of the terminal leads to the recorder lost contact. As a result, a "no feed flow" signal was sensed by the Reactor Water Level Control system causing the operating Reactor Feedwater Pump (RFP) to increase from approximately 1.98 mlb/hour flow to 5.63 mlb/hour. This flow demand exceeded the capacity of the one Condensate and Condensate Booster Pump in service; consequently, the Booster Pump discharge pressure (RFP suction pressure) decreased to less than the RFP low suction pressure trip setting of 260 psig. Following a time delay of approximately 7 seconds (the nominal time delay relay is set at 10 seconds), the RFP tripped. Reactor Vessel water level, then at 51 inches as monitored on the narrow range recorder, began to decrease. Efforts were made to restart the 1A RFP and to start the 1B RFP. However, prior to reaching the point in the RFP startup sequence where either pump was able to inject feedwater to the vessel, Reactor Vessel Water Level decreased to the Low Level Scram setpoint of 15 inches. Hence, at 1945, approximately one minute after the initiation of the feed flow transient, a Reactor Scram and Group Isolations 2, 3 and 6 occurred.

B. PLANT STATUS

Operating in the RUN mode at approximately 30% power with one Reactor Feed Pump and one Condensate and Condensate Booster Pump in service. At the time of this event, power escalation, subsequent to completion of the 1986 refueling/major maintenance outage, was in progress.

C. BASIS FOR REPORT

An automatic actuation of Engineered Safety Features, reportable in accordance with 10CFR50.73, paragraph (a)(2)(iv).

D. CAUSE OF EVENT

Personnel error, in that the acceptance testing program established to verify the correctness and adequacy of wiring changes made during the Detailed Control Room Design Review (DCRDR) effort did not verify recorder RFC-RF-96 lead input logic nor did it verify tightness of the terminal connections.

E. SAFETY SIGNIFICANCE

As noted in the CNS USAR, Chapter XIV, paragraph 4.2, Abnormal Operational Transients, eight parameter variations are listed as potential initiating causes of threats to the fuel and the nuclear process barrier. Those of concern which are related to this event include moderator temperature decrease, positive reactivity insertion, coolant inventory decrease and excess coolant inventory. Abnormal operational transients may be the result of single equipment failures

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			YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	OF	
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TEXT (If more space is required, use additional NRC Form 386A's) (17)

or single operator errors that can be reasonably expected. From the safety analysis perspective for this specific event, the loss of feed flow signal essentially constitutes an equipment failure, even though the cause, as noted above, is considered to be as a result of personnel error. In performing the analyses to determine the most significant operational transients, these single malfunctions or errors are applied to the various station systems with consideration for a variety of conditions to identify events that directly result in any of the listed parameter variations. Each event then is evaluated for the threat it poses to the integrity of the radioactive material barriers. Then, within the USAR, typically the most severe event of a group of similar events is described.

This event constituted a coolant inventory decrease. As noted in the USAR, Chapter XIV, paragraph 5.1, the concern is the threat to the fuel as the coolant becomes unable to maintain nucleate boiling. Several operational transients considered to be those most likely to limit operation are described. However, none are included which are related to a coolant inventory decrease.

As described in the CNS Technical Specifications, in the Bases for Section 2.1, paragraph A.2, the Reactor Water Low Level Scram setpoint is established at a level above the bottom elevation of the steam separator skirt. Through analyses, the determination has been made that a scram at that level adequately protects the fuel and the pressure barrier since MCPR remains well above the MCPR fuel cladding integrity limit in all cases. Hence, based upon the evaluation of expected operational transients specified in the USAR and the analyses performed to substantiate the Low Reactor Vessel Water Level Scram trip setting, this situation was of no safety significance.

Further, the Technical Specifications, in the Bases for Section 2.1, paragraph B, notes that the capacity of each Core Standby Cooling System (CSCS) component was established based on the low water level scram setpoint. Hence, under worst case conditions, for an event such as this (Total Loss of Feedwater Flow), sufficient protection is afforded to assure fuel clad integrity.

F. CORRECTIVE ACTION TAKEN OR PLANNED

Following the scram, which was accompanied by Group 2, 3, and 6 Isolations (Primary Containment, Reactor Water Cleanup and Secondary Containment including Standby Gas Treatment System Initiation, respectively), and automatic startup of the High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) system at -28.6 inches, as monitored by the narrow range Yarway indicators on Panel 9-5, control of reactor vessel water level was reestablished, the HPCI and RCIC systems were secured and normal scram recovery procedures were implemented.

Subsequently, the reversed leads to the 9-5 panel recorder RFC-RF-96 were corrected. Additionally, a search was conducted for other loose terminal connections which might affect safe operation of the plant. Two hundred and sixty-one terminal connections in the 9-5 panel which had been disconnected during the DCRDR design change work were checked. No discrepancies were found.



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FACILITY NAME (1)  Cooper Nuclear Station	DOCKET NUMBER (2)  0   5   0   0   0   2   9   8   8   7   -   0   0   3   -   0   0	LER NUMBER (6)			PAGE (3)	
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER		
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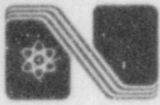
TEXT (If more space is required, use additional NRC Form 366A's) (17)

An additional two hundred and twenty-one terminal connections in the Reactor Feedwater system and HPCI control system were also checked with no discrepancies found. Of the four leads going to the Steam Flow and Feedwater Flow Recorder RFC-FR-96, two were found loose (including the loose connection which resulted in the scram). It was concluded that only the terminal leads to this recorder had not been tightened.

Further corrective action to be taken includes a review of this event by the Operations Supervisor during meetings routinely conducted with each shift crew when they are attending requalification training. Additionally, with respect to any deficiencies that may have been associated with acceptance testing or quality verification during and following completion of the Detailed Control Room Design Review activities, management direction has been given to perform a review of the acceptance testing and quality control program for this particular Design Change in an effort to assure that future installation activities are correctly accomplished and adequately verified prior to system turnover.

G. PAST SIMILAR EVENTS

While events have occurred in the past during troubleshooting and repairing various plant components, none are believed to have been initiated as a result of acceptance testing deficiencies following design change activities.



# Nebraska Public Power District

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

CNSS870067

February 6, 1987

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

Dear Sir:

Cooper Nuclear Station Licensee Event Report 87-003 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  
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