Nebraska Public Power District

COOPER NUCLEAR STATION
P.O. BOX 98, BROWNVILLE, NEBRASKA 68321
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CNSS948103

March 31, 1994

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

Dear Sir:

Cooper Nuclear Station Licensee Event Report 94-004 is forwarded as an attachment to this letter.

Sincerely,

Jerry Roque 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
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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 6

TITLE (4) Unexpected Automatic Reactor Scram Due to High Neutron Flux Caused By Partial Closure of Turbine Governor Valves Resulting From Turbine Control System Malfunction

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	02	94	94	-- 004 --	00	03	31	94	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	N
POWER LEVEL (10)	97

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)
Donald L. Reeves, Jr.	(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
X	JJ	RJX	L045	Y					

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 2, 1994, at 5:45 pm, a reactor scram from 97 percent power occurred as a result of Average Power Range Monitor (APRM) high neutron flux trip. The cause of the event was partial closure of turbine governor valves caused by a turbine digital electro-hydraulic (DEH) control system malfunction resulting in a momentary reactor pressure increase. This pressure increase caused a reduction in reactor core void fraction, resulting in reactor power reaching the APRM high flux scram setpoint. Immediately following the reactor scram, vessel inventory shrink due to void collapse resulted in reactor water level reaching the nominal Level 3 trip setting of >/= +4.5 inches with Instrument 0 being approximately 165 inches above the top of active fuel. Reactor water level was restored by the reactor feed system in combination with short term operation of both the High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) systems. The plant was stabilized in a safe condition following the transient and later placed in cold shutdown.

Through extensive failure analysis, the most likely component failure was determined to be a transistor relay on the DEH Valve Transfer Control Card. The card was replaced. During the plant startup, conducted March 12, erratic bypass valve operation occurred resulting in a small pressure spike and associated power increase. The DEH System 24 volt power supplies were determined to be the cause of the problem. Both power supplies were replaced.

A. Event Description

On March 2, 1994, at 5:45 pm, a reactor scram from 97 percent power occurred as a result of Average Power Range Monitor (APRM) high neutron flux trip. The cause of the event was partial closure of turbine governor valves caused by a turbine digital electro-hydraulic (DEH) control system malfunction resulting in a momentary reactor pressure increase. This pressure increase caused a reduction in reactor core void fraction, resulting in reactor power reaching the APRM high flux scram setpoint. Immediately following the reactor scram, vessel inventory shrink due to void collapse resulted in reactor water level reaching the nominal Level 3 trip setting of >/= +4.5 inches, with Instrument 0 being approximately 165 inches above the top of active fuel. Reactor water level was restored by the reactor feed system in combination with short term operation of both the High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) systems. The plant was stabilized following the transient in a safe condition and later placed in cold shutdown.

At the time of the scram, the only operational activity in progress was a monthly surveillance test of Diesel Generator No. 1. The first indication of the upcoming transient was the audible sound of RPS relay actuations. No prior alarm annunciations had been received. The automatic scram was followed by insertion of a manual scram. Due to lack of full in indication on control rod 10-35, the alternate rod insertion (ARI) system was manually initiated. (It was subsequently determined from the Plant Management Information System [PMIS] and scram times that all control rods, including control rod 10-35, had inserted properly.) The HPCI and RCIC systems automatically started with the lowest observed wide range level indication being approximately -21 inches. The HPCI and RCIC systems and the B Reactor Feed Pump were manually secured after verifying reactor vessel water level was being restored to the level required in the Emergency Operating Procedures. The A Reactor Feed Pump remained in service until it tripped on high reactor vessel water level. After the high level trip cleared, the RCIC pump was started manually for level control. Subsequently, the RCIC pump was secured and the A Reactor Feed Pump was restarted to maintain reactor vessel level.

Due to the reactor vessel water level transient resulting in automatic initiation of both the HPCI and RCIC systems, the Shift Supervisor at 5:58 pm elected to conservatively declare a Notification of Unusual Event (NOUE). Initial notifications to local, state and NRC officials were completed by 6:09 pm. The NOUE was maintained in effect until 9:21 pm. At 9:25 pm, State and local authorities were advised that the plant had been stabilized, all abnormal procedures exited, and the NOUE was no longer in effect. NRC was then advised of plant conditions at 9:30 pm and that the NOUE had been terminated.

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

A. Event Description (continued)

In addition to the actuation of the Reactor Protection System, this transient also initiated Primary Containment Isolation System (PCIS) Group 2, 3, and 6 Isolations at the reactor water Level 3 nominal trip setting and HPCI and RCIC system actuation at the reactor water Level 2 nominal trip setting of >/= -37 inches. The actuation of HPCI and RCIC at Level 2 for this transient is not a safety or design concern. All ESFs occurred at reactor water levels higher than the limiting settings required in Technical Specifications. All ESFs operated satisfactorily and were secured or reset manually when it was determined they were no longer needed.

B. Plant Status

Operating at 97 percent power, performing surveillance testing of Diesel Generator No. 1.

C. Basis for Report

An unexpected actuation of the RPS and ESFs, reportable in accordance with 10CFR50.73(a)(2)(iv).

D. Cause

The operational transient and ensuing automatic trip were initiated by a momentary signal from the turbine DEH System causing partial closure of turbine governor valves. A thorough inspection of the system revealed no component failures that would have resulted in the observed governor valve motion. Through extensive failure analysis and simulation, the most likely component failure was determined to be a transistor relay on the DEH Valve Transfer Control Card. Consequently, spurious actuation of the relay was postulated as the cause of the scram. The card was replaced.

During the subsequent plant startup conducted March 12, erratic bypass valve operation occurred at low power, resulting in small pressure spikes and associated power increases. Bypass valve operation was placed in manual to stabilize the plant. The monitoring system that had been installed as a result of the troubleshooting efforts for the scram identified the 24 volt power supplies to the automatic pressure control portion of the DEH System to be the likely cause of the problem. While the evaluation was in progress, the primary supply degraded notably. The secondary supply was also found degraded and could not support the system. Consequently, the cause of the scram has been attributed to a momentary degradation of the 24 volt power supplies.

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

E. Safety Significance

The March 2 event is classified as an increasing reactor pressure abnormal operational transient. The limiting increasing reactor pressure abnormal operational transient analyzed in USAR Section XIV includes the turbine trip without bypass event for MCPR and the MSIV closure with flux scram event for vessel overpressurization. The comparison of the graphs of these analyzed transients with the data from the March 2 event indicates the March 2 event was bounded (less severe in regards to safety limits) by the limiting abnormal operational transients. In the March 2 event, reactor power peaked at a much lower level than the limiting MCPR transient of load rejection without bypass. Also, reactor pressure peaked at a much lower level than predicted by GE analysis of the MSIV closure with flux trip event.

F. Safety Implications

The operational transient is most severe with the plant at full power.

G. Corrective Action

A DEH System failure was immediately apparent; therefore, a series of troubleshooting tests was initiated to identify and correct the cause of the malfunction. The objective of these tests was to determine the circuits which would have caused the observed governor valve motion, determine the circuits which could not have caused the observed governor valve motion, localize the failed components on a specific DEH System card, and ensure no other failed components existed in the DEH System. Key information from the transient was that turbine governor valves GV1, GV3, and GV4 moved simultaneously rather than sequentially (valve GV2 was closed at the time) and the turbine bypass valves did not open.

The troubleshooting tests determined that the signal causing the governor valve motion most likely came from a relay on the DEH Valve Transfer Control Card. The purpose of this relay is, upon receipt of a "not" Auto Stop Latch (ASL) logic signal indicating a main turbine trip, to send a signal which will close all governor valves simultaneously. The ASL signal circuit itself was determined not to be the cause of the valve motion observed in the transient since it, in addition to feeding the relay on this card, is also fed to other DEH System cards and would have opened the turbine bypass valves. Troubleshooting on the DEH Valve Transfer Control card could not specifically identify the component failure which would have caused the relay to generate the governor valve closure signal. However, PMIS data from the transient indicates that the failure was only momentary since the governor valves reopened. Therefore, as a conservative action, a decision was made to replace the card.

G. Corrective Action (continued)

Subsequent investigation of the conditions observed during plant startup on March 12 revealed the 24 volt power supplies to be defective. Both the primary and secondary power supplies were replaced with similar units. Both of these power supplies had been monitored during troubleshooting performed immediately after the scram and were considered to be functioning satisfactorily. An evaluation is in progress to identify long term corrective action to prevent recurrence.

In addition to the extensive investigation and troubleshooting of the DEH System, an assessment of the level transient including the differences in indicated level between the narrow range and wide range level instrument systems and an evaluation of the performance of the Reactor Feedwater Pump Control System were conducted. The results of these evaluations are as follows:

During the transient, the response of the Narrow Range (NR) reactor water level instruments varied from that of the Wide Range (WR) water level instruments. The WR instruments (as recorded by PMIS) show that a minimum reactor water level of -25 inches occurred at approximately 14 seconds after the scram. Vessel level then increased and reached a maximum value of approximately +65 inches 40 seconds later. NR instruments provided a much more varied indication. Based on one of the three channels, the indicated NR level initially decreased to about 10 inches, increased to 14 inches, and then decreased to 0 inches with a momentary spike to 3 inches. As reactor vessel level recovered, the NR level indication began to increase, eventually leveling out at approximately 60 inches.

General Electric advised that differences in the response of the NR and WR level instruments result from the differences in the location of the variable leg nozzle taps. When steam bubbles form above the level of the tap and displace water upward, there is no change in the amount of water mass above the tap and level indication does not change. However, if bubbles form in the water below the variable leg tap, then the displaced water increases the amount of water mass above the tap. As a result, the indicated water level would increase due to the presence of a steam bubble below the variable leg tap level.

In the March 2 event, the initial increase in pressure along with the reactor scram, caused a reduction in void fraction which resulted in a level decrease. Then, as pressure decreased due to the reactor scram and the turbine governor valves reopening, water began to flash into steam. A portion of the flashing was located below the NR level tap, but well above the WR variable leg tap. As a result, the pressure drop caused a momentary indicated level increase in the NR level instruments. The NR indication "bottoms" out near the NR instrument tap. Therefore, it does not show the lowest part of the transient while level is below the instrument tap (i.e., less than 0 inches).

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LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
COOPER NUCLEAR STATION	05000298	94	-- 004 --	00	6 OF 6

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G. Corrective Action (continued)

This response of the NR level instruments has been predicted in a GE analysis of a single MSIV closure event for a BWR of the same general size as CNS. The single MSIV closure event produces a pressure increase which results in a high flux trip, so the events are similar. As such, the response of the NR reactor water level instrument was an expected response for this transient.

During the scram, both reactor feed pump Lovejoy controllers went into a "track and hold" mode. The track and hold signal was generated when the selected NR water level instrument B output lowered below the low signal setpoint of alarm unit RFC-LA-121B (see GE drawing 791E257 sheet 4). Although the reactor feed pumps "locked up" in the track and hold mode, they were delivering full flow to the reactor vessel which complicated scram recovery actions taken by the operators. A special test was performed to check the level transmitter output and loss of signal setpoints. Based upon the results of the test data, the loss of signal settings for RFC-LA-121A and 121B were lowered from approximately 9.5 ma to 4 ma. The lower setting is outside the range of output from the RFC reactor water level instruments and would cause a lock and hold signal only upon a true loss of signal.

Finally, the initial NOUE notification message contained two errors. The message stated that the HPCI and RCIC systems had initiated at a reactor water level of -37 inches (the CNS Technical Specifications limiting setting for Level 2). Upon further evaluation of the event, the systems apparently initiated at a wide range level of -25 inches. The message also stated 16,000 gallons of water had been injected into the vessel by the HPCI and RCIC systems. An evaluation of system run times in conjunction with the change in emergency condensate storage tank level indication found that a more accurate value of the initial water volume injected was 700 gallons. As corrective action, revised information was provided to the NRC Duty Officer on March 5 at 6:02 pm.

H. Similar Events

None