Nebraska Public Power District

COOPER NUCLEAR STATION
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CNSS923560

March 4, 1992

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

Dear Sir:

Cooper Nuclear Station Licensee Event Report 92-002, Revision 0, is being forwarded as an attachment to this letter.

Sincerely,

Gene Peterson

FOR J. M. Meacham
Division Manager of
Nuclear Operations
Cooper Nuclear Station

JMM/bjs

Attachment

cc: R. D. Martin
G. R. Horn
R. E. Wilbur
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090067

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NRC Form 366 (9-83)

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

LICENSEE EVENT REPORT (LER)

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TITLE (4) Reactor Vessel Water Level Setpoint Inaccuracy Resulting From Reference Leg Temperature Effects That Had Not Been Correctly Addressed

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)
0 2	0 3	9 2	9 2	— 0 0 2 —	0 0	0 3	0 4	9 2		0 5 0 0 0
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OPERATING MODE (9): N

POWER LEVEL (10): 1 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)	X 50.73(a)(2)(v)	73.71(c)
20.405(a)(1)(ii)	50.36(c)(2)	X 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	
Donald L. Reeves, Jr.	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

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 February 3 and 4, 1992, Technical Specification Limiting Conditions for Operation (LCOs) were entered, requiring plant load reduction and shutdown due to discovery of non-conservative Reactor Vessel Water Level 1 nominal instrument setpoints. For Reactor Vessel Water Level instruments NBI-LIS-72A,B,C and D, this determination was made upon reviewing setpoint calculations performed in 1981 in response to General Electric (GE) Service Information Letter (SIL) Number 299, "High Drywell Temperature Effect on Reactor Vessel Water Level Instrumentation". For Reactor Vessel Water Level instruments NBI-LIS-57A and B and 58A and B, this determination was made upon reviewing the Setpoint Change Request issued in 1983 in conjunction with a Mark I Containment Program design change. The functions of the affected instruments are to initiate low pressure Emergency Core Cooling Systems, start the Diesel Generators, satisfy a portion of the Automatic Depressurization System logic and initiate the Group 1 and 7 Isolations. The reviews were performed as a result of issuance of SIL 299, Supplement 2 by GE. When these discrepancies were found on February 3, the plant was at full power; on February 4, at approximately 80 percent power.

The revised instrument setpoints were implemented within the respective LCO timeframes, eliminating the required shutdowns and allowing a return to power. The cause of the deficient setpoints was attributed primarily to personnel error on the part of NPPD employees in 1981 and 1983. The existing procedural controls are expected to eliminate errors of the type that were found. A contributing cause included the lack of a formalized setpoint procedure in 1981 and 1983 when the setpoints had last been established. This deficiency no longer exists. In addition, a communications error by GE occurred. GE will issue another supplement to SIL 299, advising of its applicability to the Group 1 and 7 Isolation functions.

A. Event Description

On January 20, 1992, an advance copy of supplement 2 to GE SIL 299 was received. The purpose of the supplement was to notify BWR owners that the information in SIL 299, dated July 25, 1979, had been misinterpreted by one utility and was potentially subject to misinterpretation by others. A clarification of the information was provided, along with a recommendation that a check of level instrument setpoint calculations be conducted.

An evaluation was performed which determined that although SIL 299 had been reviewed and properly considered when it was originally received, incorrect initial conditions and incomplete calculations for the Reactor Water Level 1 (-145.5 inches) setpoint, prescribed in the CNS Technical Specifications, resulted in a non-conservative value. This calculation performed in 1981 led to implementation of a non-conservative setpoint for Reactor Vessel Water Level instruments NBI-LIS-72A,B,C, and D. A new calculation (NEDC 92-010) was performed, based on information from SIL 299, and a new setpoint was calculated. The setpoint calculation confirmed that the existing nominal setpoint of -118.5 inches H_2O (indicated) was non-conservative and could have been below the Technical Specification value (-145.5 inches) by 15.34 inches during accident conditions (e.g., LOCA) that would result in high drywell temperatures.

On February 3, 1992, at 12 noon, due to the non-conservative setpoints, the following actions were taken:

1) The associated level instruments, NBI-LIS-72A, B, C, and D, were declared inoperable,

2) A plant shutdown, specified in the Technical Specifications in Definition 1.0.J, Limiting Conditions for Operation, was initiated,

3) Readjustment of the level instrument setpoints in accordance with Setpoint Change Request 92-09 was initiated, and

4) The need for a Temporary Waiver of Compliance was discussed with the NRC due to the uncertainty associated with the amount of time needed to readjust and verify the new instrument setpoint.

A. Event Description (Continued)

During discussions with the NRC regarding the need for a Temporary Waiver of Compliance, the acceptability of other similar Reactor Vessel Water Level instrument (NBI-LIS-57A and B and 58A and B) setpoints was discussed. The District committed to reviewing the setpoints for those instruments within twenty-four (24) hours to determine if a similar condition existed. Upon performing the evaluation, it was determined that the instrument inaccuracy caused by reference leg heating that could occur as a result of a LOCA had not been incorporated into the Setpoint Change Request issued in 1983 for these instruments when the Group 1 and 7 reactor vessel water level setpoints were reduced from ≥ -37 inches (Level 2) to ≥ - 145.5 inches. Thus, a non-conservative setpoint had also been implemented. A new setpoint calculation (NEDC 92-016) was performed based on the information from SIL 299, and two new setpoint change requests (92-010 for NBI-LIS-57A and B and 92-011 for NBI-LIS-58A and B) were initiated. The calculation confirmed that the existing nominal setpoint of -138.0 inches H_2O (indicated) was non-conservative and could also have been below the Technical Specification value (-145.5 inches) during accident conditions (e.g., LOCA) that resulted in high drywell temperatures.

On February 4, 1992, at 11:00 a.m., due to the non-conservative setpoints, the following actions were taken:

1) The associated level instruments, NBI-LIS-57A and B and NBI-LIS-58A and B, were declared inoperable,

2) A plant shutdown, in accordance with Note 2.A of Table 3.2.A of the Technical Specifications was initiated, and

3) Readjustment of the level instrument setpoints in accordance with Setpoint Change Requests 92-010 and 92-011 was initiated.

B. Plant Status

On February 3, 1992, prior to the initial event associated with level instruments NBI-LIS-72A,B,C, and D, the plant was in operation at full power. At 1:00 p.m., a load reduction was commenced to achieve Hot Shutdown within six hours as prescribed by Technical Specifications. By 5:00 p.m., the setpoint for three of the four instruments had been readjusted, and shortly thereafter, the Technical Specification Hot Shutdown LCO was exited. The setpoint for the fourth instrument was readjusted by 5:40 p.m. Approximately one half hour later, a return to full power commenced.

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

B. Plant Status (Continued)

P wer had been restored to approximately 80 percent when at 11:00 a.m., on February 4, 1992, the setpoint deficiency with level instruments NBI-LIS-57A and B and 58A and B was confirmed. Preparations for a plant shutdown to achieve Cold Shutdown in 24 hours in accordance with Technical Specification requirements were made, and within an hour, the second load reduction commenced. By 4:30 p.m., the setpoint for all four instruments had been readjusted. At 6:10 p.m., following confirmation that all required actions had been completed, the Technical Specification LCO was exited and preparations were made to return the plant to full power operation.

C Basis for Report

The non-conservative setpoint deficiency for all eight of these instruments was determined to be a condition prohibited by Technical Specifications, reportable in accordance with 10CFR50.73(a)(2)(i)(B). Additionally, under the conditions specified in SIL 299, the lowest indicated Reactor Vessel Water Level could have been only -114 inches. Since the existing setpoints for all eight level instruments were lower than this value, the setpoint deficiency is considered to be reportable in accordance with the following additional criteria:

10CFR50.73(a)(2)(ii), an event or condition that resulted in the plant including its principal safety barriers being seriously degraded (lack of automatic initiation of the Group 1 and 7 isolations), and

10CFR50.73(a)(2)(v)(C) and (D) and 10CFR50.73(a)(2)(vii)(C) and (D), an event or condition that alone could have prevented the fulfillment of the safety function of structures or systems needed to control the release of radioactive materials and mitigate the consequences of an accident.

D. Cause

The principal root cause for the setpoint deficiency is personnel error. At the time when the setpoint change request for level instruments NBI-LIS-72A,B,C, and D was initiated, an informal calculation was generated. This informal calculation contained various errors associated with the values used to formulate the setpoint. In the case of level instruments NBI-LIS-57A and B and 58A and B, the personnel involved should have compared the new setpoint to the other Level 1 setpoint, and resolved the apparent discrepancy.

D. Cause (Continued)

A contributing root cause for the setpoint deficiency is procedure deficiency. In the case of level instruments NBI-LIS-72A,B,C, and D, when the setpoint was first evaluated for the effects of high drywell temperature, the setpoint methodology did not include any requirements for design basis review. With regard to the setpoint for level instruments NBI-LIS-57A and B and 58A and B, had a formal setpoint calculation methodology existed, a design basis review would have resulted in determining that the guidance provided in SIL 299 was applicable.

Finally, with regard to the setpoint deficiency associated with level instruments NBI-LIS-57A and B and 58A and B, an additional contributing root cause (communications deficiency) has been assigned. This cause is considered appropriate due to the lack of communications from General Electric concerning the applicability of SIL 299 to the Group 1 and 7 Isolation instruments setpoint change from Level 2 ($\geq$ -37 inches) to Level 1 ($\geq$ -145.5 inches), associated with Mark I Containment changes.

E. Safety Significance

Large changes in drywell temperature, such as those associated with a LOCA, can result in differences between the measured (indicated) and actual reactor vessel water level. This issue was originally described in General Electric's Service Information Letter (SIL) No. 299, "High Drywell Temperature Effect on Reactor Vessel Water Level Instrumentation", issued July 25, 1979. With respect to Yarway water columns used at CNS, the actual level indication changes due to increasing temperature occur rather slowly, because the thermal constant of the Yarway reference leg is calculated to be 20 to 30 minutes.

With regard to level instruments NBI-LIS-72A,B,C, and D, the Reactor Water Level 1 (Low-Low-Low) initiates the Residual Heat Removal (RHR) System in the Low Pressure Coolant Injection (LPCI) mode of operation, initiates Core Spray (CS), satisfies a portion of the Automatic Depressurization System (ADS) logic, and starts the standby Diesel Generators (DGs). The Diesel Generators are started to ensure that the Emergency Core Cooling Systems (ECCSs) are available during a coincident Loss of Off-Site Power. The trip signals for Reactor Water Level 1 are set high enough to allow time for the low pressure core flooding systems to actuate or the reactor vessel to depressurize, if necessary, by actuation of ADS. The above mentioned Emergency Core Cooling Systems are used to insure that fuel cladding integrity is maintained under postulated Loss-of-Coolant Accident (LOCA) conditions.

E. Safety Significance (Continued)

The Reactor Water Level 1 signal from NBI-LIS-57A and B and 58A and B initiates closure of the Main Steam Isolation Valves (MSIVs) and Main Steam Line Drain Valves (Group 1) and closure of the Reactor Water Sample Valves (Group 7). These isolations function to maintain coolant inventory in the reactor vessel and limit the release of radioactive material in the event of gross fuel failure.

The drywell temperature related events of concern are small line breaks in the drywell. Analysis of those events shows that the diverse parameter, high drywell pressure, will initiate the RHR System in the LPCI mode of operation and the CS System before Reactor Water Level 2 (Low-Low) or Level 1 is reached. It should be noted that these low pressure systems of ECCS cannot inject into the reactor vessel until reactor pressure is below their system operating pressures (injection valve interlock is 400 psig).

The high drywell pressure signal will also initiate the DGs and the High Pressure Coolant Injection (HPCI) System. It should also be noted that the Reactor Core Isolation Cooling (RCIC) System would automatically initiate early in the LOCA scenario. This system is initiated at Level 2 which is not affected by high drywell temperature because Low-Low Level is reached before significant drywell heatup occurs.

The underlying concern addressed by SIL 299 is that the actual water level may fall below the lower instrument line tap when the high drywell temperature condition exists. If the water level falls below this tap the connected instruments will not sense further level decreases. At CNS, the control room indicator would show a level of approximately -114 inches under the conditions postulated in SIL 299. Therefore, there exists the potential that the level trips will not occur due to the effect that high drywell temperatures could have on the Yarway level instruments.

As specified in the Technical Specifications, the Reactor Water Level 1 trip signals generated by NBI-LIS-72A,B,C, and D are to be set $\geq$ 19 inches above the top of active fuel (TAF), or $\geq$-145.5 inches indicated level. In the transient analyses for the reload licensing submittal (Supplement Reload Licensing Submittal for Cooper Nuclear Station, Reload 14, Cycle 15), CNS is analyzed for a reactor vessel water level initiation signal as low as 0 inches above TAF. Due to the fact, however, that under the conditions postulated by the SIL, indicated reactor vessel water level might not decrease to the level setpoint of -118.5 inches (indicated) that existed, automatic actuation of ADS would not have been assured.

E. Safety Significance (Continued)

The Reactor Water Level 1 Trip signals generated by NBI-LIS-57A and B and 58A and B, are also specified by the Technical Specifications to be set ≥ 19 inches above TAF, or ≥-145.5 inches indicated level. General Electric document NEDE-22197 "Low-Low Set Relief Logic System and Lower MSIV Water Level Trip for CNS", justified lowering the Group 1 and 7 Isolation setpoints on Low Reactor Water Level, from Level 2 to Level 1. As previously noted, however, the level instruments setpoint had not been adjusted for the effects of drywell heating when the setpoint had been changed. This was because the concern expressed in SIL 299 had not been recognized as being applicable. With the level setpoint of -138 inches (indicated) that existed, automatic actuation of the Group 1 and 7 Isolations would not have been assured.

F. Safety Implications

Per SIL 299 the reference leg heating concerns are only relative to a small break LOCA inside containment that results in drywell temperature being raised beyond the ranges for which the instruments are calibrated. The specific scenario of concern is a small break LOCA, coincident with a loss of off-site power (resulting in the loss of feedwater), where HPCI is inoperable and where RCIC either trips after initiation due, for example, to high turbine exhaust pressure (SIL 299, Supplement 1) or is also inoperable. The probability of occurrence of this particular scenario is quite low due to the multiple failures that are required (pipe break, loss of feedwater due to loss of off-site power, HPCI inoperable, and RCIC inoperable or tripped).

Since all LOCA's inside containment will result in high drywell pressure before low water level is reached, LPCI, CS, and DG will all start (HPCI assumed inoperable) during a LOCA regardless of temperature effects on water level instrumentation. While the Yarway level instrumentation will be affected by the drywell temperature increase, alternate indications of reactor water level from newer wide range water level instrumentation (NBI-LI-85A, NBI-LI-85B, NBI-LI-91A, NBI-LI-91B, NBI-LI-91C, NBI-LI-92) are available to the operator. Reference leg injection from which correct reactor water level indication can be achieved using correction factors in the Emergency Operating Procedures is provided for these instruments. Emergency Operating Procedure guidelines require automatic ADS override for all accident scenarios when the ADS timer is activated (see Safety Evaluation of "BWR Owners' Group - Emergency Procedure Guideline, Revision 4" NEDO-31331, March 1987). The slow heatup of the instrument lines would allow the operator to follow the Emergency Operating Procedures (EOPs) and manually actuate the Safety Relief Valves (SRVs), as required. Thus, plant safety would be assured even though automatic initiation of ADS had been overridden.

NRC FORM 366A (6-89) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104
EXPIRES: 4/30/92

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE RECORDS AND REPORTS MANAGEMENT BRANCH (P-530), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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F. Safety Implications (Continued)

As previously noted, entry into the EOPs will occur well before high drywell temperature affects reactor vessel level instrumentation. The EOPs also specify ensuring that all required Group Isolations have occurred. The slow heatup of the instrument lines would allow the operator to follow the EOPs and manually isolate the MSIV's, Main Steam Line Drain Valves, and Reactor Water Sample Valves, if required. As before, plant safety would also be assured even though automatic isolation of Groups 1 and 7 might not have occurred.

G. Corrective Action

As specified in Section A, Event Description, immediate corrective action was taken to comply with applicable Technical Specification LCOs and reset the level instruments to an acceptable value based upon the concerns addressed in GE SIL 299. The setpoint calculations were performed in accordance with, and consistent with, the setpoint methodology prescribed in CNS Procedure 3.26, Instrument Setpoint and Channel Error Calculation Methodology.

The existing revision of CNS Procedure 3.26, approved August 6, 1991, requires independent design basis review and/or setpoint calculations for new setpoint changes. The setpoint calculations conform with the GE Setpoint Methodology. Therefore, future personnel errors of the type that occurred are not expected. The new procedure and setpoint methodology specifically calls for possible error terms, (e.g., high temperature effects), which must either be incorporated into the new setpoint or dismissed with adequate justification.

During the February 3, 1992 discussion with NRC related to the need for a Temporary Waiver of Compliance, the District committed to review all remaining Reactor Water Level instrumentation setpoint calculations by March 4 to ensure conservative setting limits have been implemented. This review is complete and has not revealed any further Technical Specification violations associated with Reactor Vessel Water level instrument setpoints.

Finally, with regard to the communications concern, General Electric has indicated that they will issue another supplement to SIL 299 to specifically address inclusion of high drywell temperature affects on the MSIV closure setpoint for those BWR/4 plants that changed the Technical Specification for this setpoint from Level 2 to Level 1.

H. Similar Events

None.