**Nebraska Public Power District**

COOPER NUCLEAR STATION
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TELEPHONE (402) 825-3811

NLS950044

January 31, 1995

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

Dear Sir:

Cooper Nuclear Station Licensee Event Report 94-015, Supplement 2 is forwarded as an attachment to this letter.

Sincerely,

J. T. Herron
Plant Manager

/rkg

Attachment

cc: L. J. Callan
G. R. Horn
J. H. Mueller
R. G. Jones
R. A. Sessoms
K. C. Walden
INPO Records Center
NRC Resident Inspector
R. J. Singer
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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 5 |

TITLE (4) Excessive Heatup/Cooldown During RPV Stratification Events

| 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 |
| 09 | 10 | 94 | 94 | -- 015 -- | 02 | 01 | 31 | 95 | FACILITY NAME | DOCKET NUMBER |

OPERATING MODE (9): N

POWER LEVEL (10): 000

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| | 20.402(b) | | 20.405(c) | | 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) | X | 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) |
|---|---|
| Art Alford, Senior Staff Nuclear Licensing & Safety Eng. | (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 |
|---|---|---|---|---|---|---|---|---|---|
| N/A | N/A | N/A | N/A | No | | | | | |
| | | | | | | | | | |

| 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 August 4, 1994, during a diagnostic self assessment, Nebraska Public Power District (NPPD) discovered that the average rate of reactor coolant temperature change exceeded the Technical Specification LCO 3.6.A.1 requirement of 100 degrees F/hour when averaged over a one hour period during plant trips on December 14, 1993, and March 2, 1994. During the December 14, 1993 plant trip, both reactor recirculation pumps tripped resulting in a loss of forced circulation. Cooler water accumulated in the bottom head, resulting in a temperature drop of more than 100 degrees F/hour. Also, during cooldown when the rate of depressurization was increased, enhanced core boiling increased coolant circulation and resulted in the bottom head metal and coolant temperatures exceeding a 100 degrees F/hour heatup rate. The same event occurred during the March 2, 1994, plant trip. LER 94-015 described these events. The above events did not exceed Thermal and Pressurization limits as defined in Technical Specification Figures 3.6.1.a, 3.6.1.b, and 3.6.2.

On September 10, 1994, other related events were discovered to have occurred on November 20, 1979, December 19, 1993, and March 12, 1994, during plant startups. Also, events since initial plant startup were reviewed and were found to be bounded by the analyses performed on RPV components. This revises LER 94-015-01.

Per NUREG-1022, the cause of these events is attributed to Other, specifically a Technical Specification, Section 3.6.A.1, Application Error.

Plant Status

On August 4, 1994, and September 10, 1994, when this condition was discovered, the plant was in cold shutdown with no fuel handling in progress. However, the plant was at 100% power during the December 14, 1993, event, and at 97% power during the March 2, 1994, event. During the November 20, 1979, December 19, 1993, and March 12, 1994, events the plant was in the startup mode.

Event Description

December 14, 1993, Event:
On August 4, 1994, during a diagnostic self assessment, NPPD discovered that on December 14, 1993, the average rate of reactor coolant temperature change exceeded the Technical Specification LCO 3.6.A.1 requirement of 100 degrees F/hour when averaged over a one hour period during shutdown subsequent to a plant trip. At 0134 several annunciators associated with a failure of the Reactor Feedwater Pump master controller low demand and a low reactor vessel water level alarmed. Reactor Recirculation Pump B was manually tripped to lower power rapidly. Due to expected level shrink and loss of feedwater control, Reactor Pressure Vessel (RPV) level continued to lower to the Level 2 setpoint. This caused HPCI and RCIC to initiate and Reactor Recirculation Pump A to auto trip.

Due to the trip of recirculation pumps A and B, the forced circulation was lost in the RPV. Reactor recirculation pumps could not be restarted due to prestart temperature differential limits having been exceeded. As the reactor coolant was stratifying, cooldown rates on the bottom head drain, vessel bottom head, and vessel above skirt junction exceeded 100 degrees F/hour. Procedure 2.4.2.2.4, Reactor Vessel Cold Water Stratification, was entered at 0225.

At 0430 the inboard MSIVs were closed to limit the bulk cooldown rate of the RPV per General Operating Procedure 2.1.7, Scram Recovery During Power Operation - MSIVs Open. At 0811 CRD flow was lowered to 5 gpm in accordance with subsequent actions in Abnormal Procedure 2.4.2.2.4, Reactor Vessel Cold Water Stratification, to reduce thermal stratification. Direction was given to proceed to cold shutdown and at 1055 the MSIVs were opened and a cooldown rate established. This rate, based on saturation temperature for reactor pressure, was established at approximately 55 degrees F/hour over the first hour and approximately 15 degrees F/hour over the second hour. However, the reactor depressurization enhanced core boiling and increased coolant circulation in the stratified RPV. The increase in coolant circulation resulted in heatup rates on the bottom head drain and vessel bottom head exceeding 100 degrees F/hour. On December 15, 1993, at 0414 the RHR B loop was placed in shutdown cooling. Procedure 2.4.2.2.4, Reactor Vessel Cold Water Stratification, was exited at 0516. The plant entered into cold shutdown at 0610.

March 2, 1994, Event:
On August 4, 1994, during a diagnostic self assessment, NPPD discovered that the average rate of reactor coolant temperature change exceeded the Technical Specification LCO 3.6.A.1 requirement of 100 degrees F/hour when averaged over a one hour period during the plant trip on March 2, 1994. At 1747 the reactor scrammed from 97% power on high flux due to a momentary reactor pressure increase caused by a partial closure of the main turbine governor valves due to a DEH system malfunction associated with 24 volt power supplies. The scenario of this event was similar to the December 14, 1993, event described above. During this event the forced circulation in the RPV was lost, prestart temperature differential limits were exceeded and a reactor recirculation pump could not be restarted. As the reactor coolant was stratifying, the cooldown rate at the bottom head drain exceeded 100 degrees F/hour.

Acting on lessons learned from the December 14, 1993, event, operators lowered RPV pressure from approximately 600 psig to approximately 500 psig with the main turbine bypass valves to satisfy the reactor recirculation pump prestart limits. Circulation driven by the RPV depressurization restored temperature limits and Reactor Recirculation Pump A was restarted at 1848.

At 0842 on March 3, 1994, the RHR B loop was placed in shutdown cooling. At 1041 the plant entered into cold shutdown.

Other Related Events:
On September 10, 1994, during a review of LER 94-015, NPPD discovered that three previous startups had heatup rates of the reactor wall adjacent to the flange of greater than 100 degrees F/hour. Those three startups were November 20, 1979, with a heatup rate of 102 degrees F/hour, December 19, 1993, with a heatup rate of 105 degrees/hour, and March 12, 1994, with a heatup rate of 105 degrees/hour. The high heatup rates were caused by commencing reactor startup from a moderate temperature with RPV water level in the normal operating band. This allowed the upper head region to lag behind the portions of the vessel in contact with the reactor coolant until steaming formation began. Steam generation then quickly heated up the upper head region.

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

Cause

An investigation of this event revealed the following facts:

(1) During the events addressed in this LER, the operators applied the Technical Specification LCO 3.6.A.1 requirement to average coolant temperature, as it reads, during normal heatup and cooldown; the operators were not aware of the fact that it should also be applied to bottom head temperatures during off-normal conditions and non-coupled vessel metal temperatures during startups. This is a Technical Specification Application Error.

(2) Plant startup, shutdown and scram recovery procedures prompted the operator to take into account bottom head drain and reactor recirculation loop suction temperatures and differential temperature between the dome and the bottom head drain, not RPV metal temperature. This is an Inadequacy Error in the procedures.

(3) The operator training program focused on the saturated bulk average coolant temperature of the vessel. It failed to caution that the average coolant temperature in Technical Specification LCO 3.6.A.1 also means RPV metal temperature in addition to the fluid temperature. This is a Training/Instruction Error.

The root cause of this event is a Technical Specification, Section 3.6.A.1, Application Error.

Safety Significance

The RPV is designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips.

The LCO of 100 degrees F/hour establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the reactor coolant pressure boundary. Measurement of average coolant temperature alone is not an indicator of the thermal condition of the RPV. The RPV metal temperature in the bottom head region also will have to be taken into account to determine the internal stress that the RPV has been subjected to during shutdown or heatup periods. This is because of the fact that excessive cooldown may occur at the bottom of the head region of the RPV due to off-normal conditions.

Evaluation of this type of event has demonstrated that the cumulative usage factor for the bottom head is well within the allowable code, that abnormal shutdown events have had an insignificant effect on the cumulative usage factor, and that the total number of transient cycles experienced by the RPV to date are below allowable limits. It is therefore concluded that the safety significance of these abnormal shutdown and startup events is minimal.

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

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EXPIRES 5/31/95

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# 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 | 94 | -- 015 -- | 02 | 5 OF 5 |

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

Corrective Action

Submit a proposed Technical Specification change to clarify LCO 3.6.A.1, clearly indicating the applicability.

Operator Requalification Classroom and Simulator Training has been conducted concentrating on performance of actions to minimize RPV stratification with emphasis on Technical Specification heatup and cooldown temperature monitoring requirements.

A Software Design Change Request to develop and implement a dynamic computer display to calculate the required data enabling faster reactor recirculation pump recovery has been initiated.

Transient events from initial startup to present have been reviewed. The results of this review and evaluation was that the fatigue margin still remained well within the ASME Code limits. Therefore, any events in which the 100 degrees F/hour rate was exceeded, but not accounted for, is enveloped by the analyses that were performed on the most limiting RPV components. The process and methods used to assess and monitor primary system boundary integrity were investigated and actions were taken to demonstrate that adequate fatigue margins exist for continued operation during the current reload. The basis for the determination was a conservative analysis. A refined analysis will be performed to support continued operation for succeeding operating cycles. Additionally, the refined analysis is to have sufficient detail to develop operator guidance and procedural enhancements to minimize reductions in fatigue margins at the vessel component level.

Similar Events

None.

Correspondence No: NLS950044

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 |
|---|---|
| Submit a proposed Technical Specification change to clarify LCO 3.6.A.1 clearly indicating the applicability. | N/A |
| Revise primary system boundary integrity analysis by minimizing conservatism in methods and classification of plant events. | N/A |
| Develop operator guidance and procedural enhancements to minimize reduction in fatigue margins at the vessel component level. | N/A |
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