Nebraska Public Power District

COOPER NUCLEAR STATION
P.O. BOX 98, BROWNVILLE, NEBRASKA 68321
TELEPHONE (402)825-3811
FAX (402)825-5205

NLS950236

December 26, 1995

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555-0001

Dear Sir:

Cooper Nuclear Station Licensee Event Report 95-017, Supplement 1 is forwarded as an attachment to this letter.

Sincerely,

J. T. Herron
Plant Manager

CCT

Attachment

cc: L. J. Callan
G. R. Horn
J. H. Mueller
R. G. Jones
R. A. Sessoms
M. F. Peckham
R. L. Gardner
N. E. Champlin
T. N. Ferrando
INPO Records Center
NRC Resident Inspector
W. Turnbull
CNS Training
CNS Quality Assurance



IE22
1

NRC FORM 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 04/30/98



# 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)

Safety/Relief and Safety Valves Found Outside Technical Specification Limiting Safety System Setting

| 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 |
| 10 | 23 | 95 | 95 | -- 017 -- | 01 | 12 | 26 | 95 | FACILITY NAME | DOCKET NUMBER |

OPERATING MODE (9): N

POWER LEVEL (10): 0

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

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| | 20.2201(b) | | 20.2203(a)(2)(v) | | 50.73(a)(2)(i) | | 50.73(a)(2)(viii) |
| | 20.2203(a)(1) | | 20.2203(a)(3)(i) | | 50.73(a)(2)(ii) | | 50.73(a)(2)(x) |
| | 20.2203(a)(2)(i) | | 20.2203(a)(3)(ii) | | 50.73(a)(2)(iii) | | 73.71 |
| | 20.2203(a)(2)(ii) | | 20.2203(a)(4) | | 50.73(a)(2)(iv) | | OTHER |
| | 20.2203(a)(2)(iii) | | 50.36(c)(1) | | 50.73(a)(2)(v) | | Specify in Abstract below or in NRC Form 366A |
| | 20.2203(a)(2)(iv) | | 50.36(c)(2) | X | 50.73(a)(2)(vii) | | |

LICENSEE CONTACT FOR THIS LER (12)

| NAME | TELEPHONE NUMBER (Include Area Code) |
|---|---|
| Calvin C. Taylor, Licensing and Compliance Specialist | (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 | SB | RV | T020 | Y | | | | | |
| X | SB | RV | D243 | 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)

While in cold shut down for the current refueling outage, (RE16), eight Safety Relief Valves (SRVs) were removed and sent to the Westinghouse testing facility in Banning, California for testing in accordance with Cooper Nuclear Station (CNS) Technical Specifications (TS). In the period between October 24-26, 1995, four of the eight SRVs lift pressures were found higher than TS Limiting Safety System Settings tolerance of +/- 11 psi (+/- 1%). This has been a recurring problem in the industry with several failures noted at CNS as well as other nuclear facilities.

A Safety Valve was also removed and sent to the same testing facility for a TS required surveillance. On October 23, 1995, the SV lift pressure was found lower than TS Limiting Safety System Settings tolerance of +/-13 psi.

The cause of the SRV setpoint drift is attributed to corrosion bonding of the pilot disc to the pilot seat, (NUREG 1022, Appendix B, Cause Code B, Design, Manufacturing, Construction/Installation). CNS installed 0.3% platinum alloy discs in four of eight SRVs installed after testing in December 1994. CNS will continue to monitor industry efforts to resolve the corrosion bonding setpoint drift phenomena and if operation demonstrates that changing to 0.3% platinum discs in SRVs is effective, the remaining seats will be replaced in a future outage.

The suspected cause of the SV setpoint drift is valve seat leakage leading to elevated temperatures, spring relaxation and set point drift on the low side, however, no cause can be ascertained with certainty, (NUREG 1022, Appendix B, Cause Code X, Other). The causes of seat leakage can be foreign material intrusion, corrosion, seat/disc alignment, and vibration. CNS will continue to monitor industry efforts to address setpoint drift of SVs.

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

## PLANT STATUS

Cooper Nuclear Station (CNS) was in cold shutdown for the current refueling outage (RE16).

## EVENT DESCRIPTION

Eight Safety Relief Valves (SRVs) [EIIS identifier - RV] were removed and sent to the Westinghouse testing facility in Banning, California for testing in accordance with CNS Technical Specifications (TS). In the period between October 24-26, 1995, four of the eight SRVs lift pressures were found higher than TS Limiting Safety System Settings tolerance of +/- 11 psi, (1%).

In addition, one of three SVs [RV] was also tested at the same facility in accordance with CNS TS. On October 23, 1995, the RV lift pressure was found to be lower than TS Limiting Safety System Setting tolerance of +/- 13 psi, (1%). The SV failure was not reported in the original 10CFR50.73 submittal on November 24, 1995, due to an administrative oversight. The CNS system engineer was notified of the failure on October 23, 1995, by a CNS representative at the test facility, but through an administrative oversight, the system engineer failed to document this failure until November 27, 1995.

The SV and SRVs were refurbished as necessary and recertified. The results of the testing are as follows:

| Location | S/N | Set Press | As Found 1st,2nd,3rd Lifts | % Drift (Neg. value) | Test Date |
|---|---|---|---|---|---|
| MS-RV-70ARV | BL-02463 | 1240 | 1221, 1226, 1208 | (1.5), (1.1), (2.6) | 10/23/95 |
| MS-RV-71ARV | 379 | 1100 | 1297, 1104, 1099 | 17.9, 0.4, (0.1) | 10/26/95 |
| MS-RV-71BRV | 380 | 1100 | 1120, 1097, 1092 | 1.8, (0.3), (0.7) | 10/25/95 |
| MS-RV-71CRV | 385 | 1090 | 1100, 1088, 1087 | 0.9, (0.2), (0.3) | 10/25/95 |
| MS-RV-71DRV | 387* | 1080 | 1080, 1082, 1080 | none, 0.2, none | 10/23/95 |
| MS-RV-71ERV | 377* | 1090 | 1098, 1097, 1085 | 0.7, 0.6, (0.5) | 10/24/95 |
| MS-RV-71FRV | 381* | 1080 | 1106, 1074, 1072 | 2.4, (0.6), (0.7) | 10/25/95 |
| MS-RV-71GRV | 376* | 1100 | 1107, 1089, 1089 | 0.6, (1.0), (1.0) | 10/24/95 |
| MS-RV-71HRV | 378 | 1090 | 1186, 1091, 1082 | 8.8, 0.1, (0.7) | 10/24/95 |

* Denotes valves with BWROG recommended platinum stellite pilot discs

| FACILITY NAME (1) | DOCKET | LER NUMBER (6) | | | PAGE (3) |
|---|---|---|---|---|---|
| | | YEAR | SEQUENTIAL | REVISION | |
| COOPER NUCLEAR STATION | 05000298 | 95 -- | 017 -- | 01 | 3 OF 5 |

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

CAUSE

The cause of the SRV set point drift is attributed to corrosion bonding of the pilot disc to the pilot seat, (NUREG 1022, Appendix B, Cause Code B - Design, Manufacturing, Construction/Installation).

The SRVs installed at CNS are Target Rock pilot actuated valves which are typical for BWRs. Set point drift of Target Rock SRVs above their required set point tolerance of one percent has been an industry wide problem for which the BWR Owners Group (BWROG) has been actively pursuing resolution for several years.
Industry information has identified that radiolytically produced hydrogen and oxygen can concentrate in the immediate vicinity of the pilot disc and seat interface as a result of condensation of reactor steam. The BWROG concluded that the major contributor to corrosion induced upward set point drift is concentrated oxygen, which increases the electro-chemical potential of the pilot disc material. The BWROG has determined that a catalyst should be installed which would recombine the oxygen and hydrogen in the vicinity of the disc and seat interface so as to maintain the oxygen concentration below that required to facilitate corrosion. After evaluating the catalysts, the BWROG recommended replacing the Stellite 6 pilot discs in half of the SRVs with new pilot discs of Stellite 6 alloyed with 0.3% platinum.

CNS has been operated continuously from February 1995 until October 1995 after a shutdown from May 1994 to February 1995 interrupted Cycle 16. The length of that unscheduled outage made it prudent to test the SRVs in December 1994 and CNS installed the 0.3% platinum discs in four of the eight SRVs installed at that time. All eight of the SRVs installed at that time were tested during the current refueling outage.

Failure of as found set point testing for SRVs has been an industry wide problem for several years. CNS has had a failure rate above the industry average. Eight SRVs were tested in December 1994. Four of the eight were higher than their required tolerance of +/- 11 psi and one was below the tolerance. Eight SRVs were tested in 1993 with seven of the eight higher than the required tolerance. A review of previous failures has revealed no correlation between the magnitude of setpoint drift and either location or serial number.

There is insufficient evidence for the SV failure to ascertain a cause, (NUREG 1022 cause code X, Other). A primary causal factor, valve seat leakage, is widely accepted in the industry. It leads to elevated temperatures, spring relaxation and set point drift on the low side. The causes of seat leakage can be foreign material intrusion, corrosion, seat/disc alignment, and vibration. However, the vendor suggested that the seat leakage in this failure was the result of vibrations experienced during shipment from CNS to the test facility. This hypothesis could not be proven and therefore a cause cannot be determined with certainty.

The CNS Engineer overseeing the testing observed that when the valve was brought to operating temperature and pressure some leakage was observed at the disc/seat interface. The CNS Engineer stated that the valve inspection at the test facility showed no evidence of foreign material contaminating the disc/seat interface.

The SV is a model 3777QA RT22 spring loaded valve which is typical for Main Steam applications in BWRs and PWRs. Industry experience was reviewed by a search of the NPRDS database for similar failures. Approximately 70 failures were reviewed from 6 plants. Roughly 2/3 of the failures at these facilities were as-found set point below the acceptable range. The identified causes varied but the majority did refer to pre-test leakage leading to the set point drift. The cause of leakage was not typically identified.

CNS experience is consistent with the nuclear industry experience. Roughly 2/3 of the CNS failures were as-found set points below the acceptable range.

| FACILITY NAME (1) | DOCKET | LER NUMBER (6) | | | PAGE (3) |
|---|---|---|---|---|---|
| | | YEAR | SEQUENTIAL | REVISION | |
| COOPER NUCLEAR STATION | 05000298 | 95 | 017 | 01 | 4 OF 5 |

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

## SAFETY SIGNIFICANCE

General Electric (GE) reviewed the current as-found setpoints of the SRVs for possible impact on previous safety analyses. GE concluded in their evaluation that previous analyses remain applicable in that there is ample margin available to avoid any potential plant safety concerns and there is no significant safety impact in vessel over pressure margin, thermal limits, ECCS/LOCA performance, HPCI/RCIC performance, containment response, containment integrity, or steam line integrity. GE determined in the current analysis that with SRV A (serial number 379) and H (serial number 378) drifting to 1297 and 1186 psig respectively and the remaining valves assumed to be at +3% above the nominal setpoint, the calculated vessel bottom head pressure would be 1263 psig. This is higher than the peak vessel pressure reported for the Cycle 16 reload analysis (1241 psig), but well below the vessel overpressure limit of 1375 psig.

Furthermore, the calculated vessel dome pressure for the overpressurization event with drifted SRV setpoints is 1244 psig. Therefore, the complement of the SRVs with setpoints at or below approximately 1244 psig have sufficient capacity to ensure vessel pressure remains well within the 1375 psig overpressure limit.

The CNS USAR states that the Safety Design Bases of the Nuclear System Pressure Relief system is to prevent overpressurization of the nuclear system in order to prevent failure of the nuclear system process barrier due to pressure. The SV actuation setpoint within TS Section 2 Limiting Safety System Settings protects the nuclear system process barrier from failure due to pressure. The CNS USAR Safety Evaluation states that the basis for sizing the safety valves is the most severe event postulated, closure of all MSIV's with the reactor scram on high neutron flux level. The As-Found set point was less than, and more conservative than, the acceptable range for the Limiting Safety System Setting. The margin of safety has not been decreased and there is no safety significant impact in vessel overpressure margin of safety.

An analysis of a reactor shutdown by the backup high neutron flux scram with a closure of all MSIVs credits a design safety valve capacity of 15% rated flow in conjunction with a design relief valve capacity of 61% rated flow to maintain adequate margin below ASME code allowable pressure in the nuclear system.

The CNS USAR states that a Power Generation Design Bases of the Nuclear System Pressure Relief system is that the SRV's shall prevent the opening of the spring-loaded SV's during normal plant isolations and load rejections. The USAR Power Generation Evaluation evaluates a less severe event, the turbine trip without bypass as the basis for sizing the SRVs to prevent SV actuation. For normal plant isolations and load rejection events, this event represents the fastest possible steam flow shutoff and therefore represents the potential for the most severe pressure transient. For this transient, the evaluation determined a peak pressure at the safety valves is 1192 psig. With the SRV setpoint drift, GE calculated a 19 psi higher peak vessel pressure for the Cycle 16 analysis. A conservative estimate of the effect of the drifted SRV set points on the turbine trip without bypass was obtained by adding 19 psi to the USAR Power Generation Evaluated peak pressure of 1192 to conclude that peak pressure at the SVs would not exceed 1211 psig. The first As-Found automatic actuation of the SV occurred at 1221 psig and therefore would not have actuated in this scenario.

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

## CORRECTIVE ACTIONS

A CNS Special Test Procedure is controlling the evaluation and implementation of the BWROG recommendation to correct corrosion bonding setpoint drift phenomena. Three of the four SRVs with BWROG recommended stellite platinum alloy discs were within +/- 11 psi of their set pressure. This success rate appears to be an improvement over past performance. Also, CNS is in the process of converting to standard Tech Specs (NUREG 1433) which will allow an SRV setpoint tolerance of +/- 33 psi. All four SRVs with stellite platinum discs were within this tolerance.

The SRV surveillance procedure has been revised to include notifying the system engineer, shift supervisor, IST engineer, and Nuclear Licensing of any failed SRV TS surveillances. The SV surveillance procedure will be revised for the same notifications.

1. CNS will continue to monitor industry efforts to resolve the corrosion bonding setpoint drift phenomena. (As committed to in CNS LER 93-013)

2. If operation demonstrates 0.3% platinum discs in SRVs is effective, the remaining seats will be replaced in a future outage. (As committed to in CNS LER 94-033)

3. CNS will continue to monitor industry efforts to resolve setpoint drift of the Safety Valves.

4. For future SRV/SV tests at off-site test facilities, the CNS representative will complete the surveillance procedure at the test facility and make appropriate notifications as specified in the surveillance procedure.

## PREVIOUS EVENTS

LER 94 033 Safety Relief Valve Setpoint Variance Not Within Technical Specification Limits

LER 93-013 Safety/Relief and Safety Valve Setpoint Variance Not Within Technical Specification Limits

LER 91-015 Safety/Relief and Safety Valve Setpoint Variance Not Within Technical Specification Limits

LER 90-003 Safety/Relief and Safety Valve Setpoint Variance Not Within Technical Specification Limits

LER 89-015 Safety/Relief Valve Setpoint Variance Not Within Technical Specification Limits

LER 88-009 Setpoint Variance and Operability Concerns Associated With Safety Relief Valves Discovered During Surveillance Testing

LER 86-032 Main Steam Safety Relief Valve Setpoint Drift and Stuck Pilot Valve Inoperability Discovered During Scheduled Valve Testing and Refurbishment

LER 85-003 Setpoint Drift of Safety and Safety Relief Valves

Correspondence No: NLS950236

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 |
|---|---|
| CNS will continue to monitor industry efforts to resolve the corrosion bonding setpoint drift phenomena. | Ongoing until appropriate resolution determined and successfully implemented. |
| If operation demonstrates 0.3% platinum discs in SRVs is effective, the remaining seats will be replaced in a future outage. | Refueling Outage RE17 |
| CNS will continue to monitor industry efforts to resolve setpoint drift of the Safety Valves. | Ongoing until appropriate resolution determined and successfully implemented. |
| For future SRV/SV tests at off-site test facilities, the CNS representative will complete the surveillance procedure at the test facility and make appropriate notifications as specified in the surveillance procedure. | Ongoing |
| | |
| | |
| | |
| | |
| | |
| | |
| | |
| | |
| | |