NRC Form 366
(9-83)

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

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Cooper Nuclear Station	0 5 0 0 0 2 9 8	1 OF 0 5

TITLE (4) Setpoint Variance and Operability Concerns Associated with Safety Relief Valves and Safety Valves Discovered During Surveillance Testing

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 5 0 0 0
0 4	0 7	8 8	8 8	— 0 0 9 —	0 0	0 5	0 6	8 8		0 5 0 0 0

OPERATING MODE (9): N

POWER LEVEL (10): 0 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)	50.73(a)(2)(v)	73.71(c)
20.405(a)(1)(ii)	50.36(c)(2)	50.73(a)(2)(vii)	OTHER (Specify in Abstract below and in Text, NRC Form 366A)
20.405(a)(1)(iii)	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
X	S B	R V	D 2 4 5	Y					
X	S B	R V	T 0 2 0	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 fifteen single-space typewritten lines) (16)

During performance of Safety Relief Valve (SRV) and Safety Valve (SV) testing required by the CNS Technical Specifications, four (4) of the eight (8) installed SRVs and two (2) of the three (3) installed SVs were sent to Wyle Laboratories in Huntsville, Alabama, to be bench checked. Three problems were discovered as follows:

1) One SRV which should have lifted at 1100 psig ± 11 psi actually lifted at 1220 psig,
2) One SRV set to actuate at 1080 psig ± 11 psi could not be tested as received due to a pilot assembly to main body flange leak. The pilot assembly was removed and separately tested with satisfactory results, and
3) One SV which should have lifted at 1240 psig ± 13 psi actually lifted at 1268 psig.

Bench checks conducted on the other three valves were satisfactory.

The valves were inspected and refurbished in accordance with standard procedures employed by Wyle Laboratories and the respective valve manufacturers and subsequently tested satisfactorily. The inspections conducted proved to be inconclusive from the perspective of identifying the cause of the observed setpoint variance.

These discrepancies were evaluated by General Electric and determined to have no safety significance.

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A. Event Description

In accordance with requirements prescribed in paragraph 4.6.D.1 of the CNS Technical Specifications, four (4) of the eight (8) installed 6 x 10 Pilot-Operated Safety Relief Valves (SRVs), Part No. 7567F, manufactured by Target Rock, and two (2) of the three (3) installed Main Steam Safety Valves (SVs), Part No. 3777 QA-RT22, manufactured by Dresser Industries, were removed from their installed positions and transported to an offsite facility (Wyle Laboratories, Huntsville, Alabama) to conduct bench checks and, as required by paragraph 4.6.D.2, to disassemble and inspect at least one (1) SRV. The following discrepancies were discovered:

1) The set pressure for one SRV, MS-RV-71-BRV, was determined to be 1220 psig. This is one of three (3) SRVs for which the setpoint is to be 1100 psig ± 11 psi in accordance with paragraph 2.2.1.B of the CNS Technical Specifications.

2) The set pressure for another SRV, MS-RV-71-FRV, could not be determined in its as-received condition due to a gasket leak at the pilot assembly to main valve body flange. Water leakage from this flange had been noted before the valve was removed from its installed location when the Main Steam Lines were filled to facilitate Main Steam Line Plug installation. This is one of two (2) SRVs for which the setpoint is to be 1080 psig ±11 psi. The pilot assembly from this valve was removed and installed in a slave base and body which were inspected and evaluated by Wyle Laboratories personnel to be acceptable for test purposes. A successful bench check was then conducted and the set pressure during four runs was determined to be between 1078 and 1090 psig.

3) The set pressure for one SV, MS-RV-70-BRV, was determined to be 1268 psig. This is one of three (3) safety valves for which the setpoint is to be 1240 psig ±13 psi.

B. Plant Status

Shutdown for the 1988 Refueling outage which commenced March 5, 1988.

C. Basis for Report

The two (2) deficiencies associated with setpoint variance beyond the allowable tolerance potentially could have resulted in the plant being in a condition outside the design basis, reportable in accordance with 10CFR50.73(a)(2)(ii). The one SRV that could not be tested in the as-received condition is being reported as an item of interest.

D. Cause

With regard to the two valves exhibiting setpoint variance (one safety relief valve and one safety valve), no specific deficiencies were noted during subsequent valve inspection and refurbishment that would enable a conclusion to be drawn regarding cause. It should be noted that during testing in the as-received condition, multiple runs were made; four for the SRV and three for the SV. The setpoint variance beyond the allowable tolerance (±11 psi for the SRV; ±13 psi for the SV) was only identified on the first run. The setpoints determined on the second and successive runs in each case were within the allowable tolerance.

The flange leakage problem which was observed on the one SRV was apparently due to a gasket malfunction. Upon being appraised of the flange leakage problem which precluded as-received testing, an investigation was conducted in an effort to determine if there was any evidence or indications of leakage from this valve during the past operating cycle and, if so, when such leakage may have started. Since leakage from this flange during operation would be to the Drywell atmosphere, Reactor Coolant System (RCS) leakage data for the last cycle was reviewed. During the period from January 1, 1988, to January 26, 1988, the leakage trended upward from approximately .03 gpm (15 lbm/hr) to .25 gpm (125 lbm/hr). Leakage during the remaining portion of the cycle was nearly constant. A review of charts of temperature recorders PC-TR-500A, B, C, and D (Drywell fan coil units inlet and outlet air temperatures) and PC-TR-102 and 103 (Reactor Recirculation pump A and B area temperatures) for January through March 1988 showed no indications of steam leakage. Examination of the insulation set which surrounds the SRV for signs of steam leakage revealed no staining, warping or discoloration. Wyle Laboratories personnel and General Electric representatives who examined the valve stated that the base to body sealing surfaces showed no evidence of steam cutting. The research described above indicates that valve leakage during plant operation was insignificant and, therefore, the valve would have operated as designed.

E. Safety Significance

General Electric was contacted and requested to evaluate the potential impact upon the most severe pressurization transients as a result of the SRV and SV deficiencies discovered. As previously noted, due to the flange leakage experienced by one of the SRVs, it was not possible to perform set pressure testing in its as-received condition. The leakage from this valve was subsequently found to have been due to a gasket failure. However, because of the absence of damage to the gasket seating surface and lack of other leakage indications, it is expected that the gasket was intact throughout the operating cycle. Additionally, a separate test of the pilot assembly satisfactorily demonstrated its operability. Further, as regards operability of this valve, if it were conservatively assumed that all reactor leakage within the primary containment was attributable to the SRV, this leakage would be, at most, 125 lbm/hr. Leakage of this magnitude would not be expected to significantly affect setpoint.

For purposes of the analysis conducted by General Electric, it was conservatively assumed that this SRV was inoperable. With regard to the seven operable valves, an average SRV drift of less than 4% was conservatively determined by assuming:

1) Two SRVs would have actuated at the "as-received" test setpoint (1220 psig) of the single valve which exceeded the Technical Specification limit (although only one SRV drifted, two SRVs are assumed to have drifted since only four of the eight in-service valves were tested).

2) The remaining five SRVs would actuate at the Technical Specification upper limit (nominal setpoint +11 psi).

The average SV drift was conservatively determined to be less than 2% by assuming:

1) Two SVs would have actuated at 1268 psig (the "as-received" data for one of the two valves tested).

2) One SV would actuate at the Technical Specification upper limit (1240 psig + 13 psi).

Based on previous studies performed by General Electric, had the limiting overpressurization transient actually occurred with one SRV inoperable and an upward setpoint drift of 4% on the SRVs and 2% on the SV, the maximum reactor pressure would have been well below the ASME code limit of 1375 psig. The previously performed studies also indicate that no significant change in the LOCA response for CNS will occur.

From the Cycle 11 reload licensing submittal for CNS (the past operating cycle), the two most limiting events in terms of minimum critical power ratio (MCPR) are the load rejection without bypass and the feedwater controller failure - maximum demand events. In each of these events, the SRVs do not actuate until after the MCPR occurs. Consequently, a delayed SRV actuation due to upward setpoint drift would not impact the CNS MCPR operating limit.

One postulated operational concern associated with SRV setpoint drift stems from the potential for higher steam line pressures to be reached during postulated operational transients. The most limiting event in this category is the load rejection without bypass event. The Cycle 11 evaluation of this event indicates that the peak steam line pressure is below the nominal SV setpoint by 61 psi. Consequently, the unpiped SVs

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

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

would not actuate, even if their setpoints were to drift downward by the allowable 13 psi. If this event is postulated with one SRV inoperable and an upward drift of 4% on the SRVs, the maximum steam line pressure is not expected to reach the SV setpoints, even in the event that one of the SV setpoints drifts downward by the allowable 13 psi.

In conclusion, the upward drift of the SRV and SV setpoints, and potentially having one inoperable SRV, would not have impacted any plant safety limits. Consequently, safe plant operation was not compromised.

F. Corrective Action

The SRV that could not be tested in its as-received condition was disassembled by personnel from Target Rock with assistance by Wyle. The components were inspected, cleaned, and machined as necessary and the flange gasket, previously found defective, was replaced. The valve was reassembled and four (4) successful set pressure runs were made. The pilots for the other three (3) SRVs were also disassembled, inspected, cleaned and reassembled. Subsequent testing, which included four (4) runs on each valve, was accomplished satisfactorily.

With regard to the Safety Valves, both were disassembled and inspected by personnel from Dresser Field Services with support from Wyle. Due to evidence of pitting/minor steam cutting, the valve discs were replaced. Insufficient material existed from prior maintenance efforts to properly lap the surfaces. Similar problems were noted with the valve seats; however, the irregularities were removed by lapping since adequate material depth remained. After reassembly the valves were successfully tested with three (3) runs made on each valve.

G. Prior Similar Events

Deficiencies observed during prior testing have been observed and reported, the two most recent being:

LER 86-032, transmitted December 10, 1986

LER 85-003, transmitted July 3, 1985, with a followup report transmitted August 19, 1985.