10 CFR 50.73

CP&L
A Progress Energy Company

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United States Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261/LICENSE NO. DPR-23

LICENSEE EVENT REPORT NO. 2002-001-00
FOUR MAIN STEAM SAFETY VALVES FAIL TO MEET
ACCEPTANCE CRITERIA DURING LIFT PRESSURE TESTING

Ladies and Gentlemen:

The attached Licensee Event Report is submitted in accordance with the requirements of 10 CFR 50.73. Should you have any questions regarding this matter, please contact Mr. C. T. Baucom.

Sincerely,

Timothy P. Cleary

Timothy P. Cleary
Plant General Manager

CAC/cac

Attachment

c: Mr. L. A. Reyes, NRC, Region II
Mr. R. Subbaratnam, NRC, NRR
NRC Resident Inspector, HBRSEP

IE22

NRC FORM 366 (1-2001) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
H. B. Robinson Steam Electric Plant, Unit No. 2	05000261	1 OF 8

TITLE (4)

Four Main Steam Safety Valves Fail to Meet Acceptance Criteria During Lift Pressure Testing

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MO	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
10	09	2002	2002	- 001	- 00	12	09	2002	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9): 1

POWER LEVEL (10): 93%

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply) (11)

	20 2201(b)		20.2203(a)(3)(ii)		50.73(a)(2)(ii)(B)		50.73(a)(2)(ix)(A)
	20 2201(d)		20 2203(a)(4)		50.73(a)(2)(iii)		50.73(a)(2)(x)
	20.2203(a)(1)		50 36(c)(1)(i)(A)		50.73(a)(2)(iv)(A)		73.71(a)(4)
	20.2203(a)(2)(i)		50.36(c)(1)(ii)(A)		50 73(a)(2)(v)(A)		73 71(a)(5)
	20.2203(a)(2)(ii)		50 36(c)(2)		50.73(a)(2)(v)(B)		OTHER Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iii)		50 46(a)(3)(ii)		50.73(a)(2)(v)(C)		
	20 2203(a)(2)(iv)		50 73(a)(2)(i)(A)		50 73(a)(2)(v)(D)		
	20 2203(a)(2)(v)	X	50 73(a)(2)(i)(B)		50 73(a)(2)(vii)		
	20 2203(a)(2)(vi)		50 73(a)(2)(i)(C)		50 73(a)(2)(viii)(A)		
	20 2203(a)(3)(i)		50 73(a)(2)(ii)(A)		50 73(a)(2)(viii)(B)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
C. T. Baucom	843-857-1253

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
X	SB	RV	Crosby	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)

During testing of the Main Steam Safety Valves (MSSVs) on October 9 and 10, 2002, it was determined that four of the 12 MSSVs had as-found lift pressures that exceeded the Technical Specifications (TS) tolerance of +/- 3%. The four valves' as-found lift pressures were over TS lift pressures by the following amounts: SV1-1A was 3.1% over, SV1-2B was 4.1% over, SV-1C was 3.8% over, and SV1-2C was 6.9% over. A condition report (number 73940) was initiated and the corrective action program significant adverse condition investigation has been completed. The root cause of the high lift pressures was attributed to mechanical component failure/degradation based on slight binding of the spindle on the guide bearing. The corrective actions for this event included maintenance on the MSSVs and post-maintenance testing as required for the maintenance performed. This condition was determined to be reportable in accordance with 10 CFR 50.73(a)(2)(i)(B) as a condition prohibited by the plant's TS, based on guidance contained in NUREG-1022, "Event Reporting Guidelines 10 CFR 50.72 and 50.73," Revision 2.

I. DESCRIPTION OF EVENT

On October 9, 2002, the H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2, reactor was operating at approximately 93% power. At 0829 hours, testing of the main steam system [SB] safety valves [RV] was initiated in accordance with procedure EST-028, "Main Steam Safety Valve Testing (Refueling Shutdown Interval and As Needed After Maintenance)."

By the completion of the testing on October 10, 2002, the entire set of 12 safety valves (four valves on each of the three main steam lines) had been tested. Four safety valves were found with setpoints above the test acceptance criteria.

The first safety valve test failure occurred at 0948 hours on October 9, 2002, during testing of valve SV1-2B, which is one of the four main steam system safety valves on the "B" steam generator [SG] main steam line. The applicable HBRSEP, Unit No. 2, Technical Specifications limiting condition for operation (LCO) required action for inoperability of SV1-2B had been entered at 0943 hours to accommodate the valve test. The valve was adjusted and subsequently retested. The valve was returned to operable status and the LCO action statement was exited at 1023 hours. This test failure required two additional valves, beyond the original four valves selected, to be tested in accordance with the inservice testing program.

The second safety valve test failure occurred at 1250 hours on October 9, 2002, during testing of valve SV1-2C, which is one of the four main steam system safety valves on the "C" steam generator main steam line. The applicable Technical Specifications LCO required action for inoperability of SV1-2C had been entered at 1247 hours to accommodate the valve test. The valve was adjusted and subsequently retested. The valve was returned to operable status and the LCO action statement was exited at 1327 hours. This test failure required the entire set of main steam system safety valves to be tested.

The third safety valve test failure occurred at 1412 hours on October 9, 2002, during testing of valve SV1-1C, which is one of the four main steam system safety valves on the "C" steam generator main steam line. The applicable Technical Specifications LCO required action for inoperability of SV1-1C had been entered at 1407 hours to accommodate the valve test. The valve was adjusted and subsequently retested. The valve was returned to operable status and the LCO action statement was exited at 1445 hours.

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

The fourth safety valve test failure occurred at 1123 hours on October 10, 2002, during testing of valve SV1-1A, which is one of the four main steam system safety valves on the "A" steam generator main steam line. The applicable Technical Specifications LCO required action for inoperability of SV1-1A had been entered at 1122 hours to accommodate the valve test. The valve was retested satisfactorily. The valve was returned to operable status and the LCO action statement was exited at 1145 hours.

A condition report (number 73940) was initiated and the corrective action program significant adverse condition investigation has been completed. This reportable event and the associated significant adverse condition investigation was reviewed by the Plant Nuclear Safety Committee on December 4, 2002.

This condition was determined to be reportable based on guidance contained in NUREG-1022, "Event Reporting Guidelines 10 CFR 50.72 and 50.73," Revision 2. Example 3 in Section 3.2.2 of NUREG-1022, Revision 2, states that multiple test failures are reportable under 10 CFR 50.73(a)(2)(i)(B), "Any operation or condition prohibited by the plant's Technical Specifications," if the existence of similar discrepancies in multiple valves is an indication that the discrepancies may well have arisen over a period of time.

HBRSEP, Unit No. 2, Technical Specifications LCO Section 3.7.1 provides the operability requirements for the Main Steam Safety Valves (MSSVs). LCO 3.7.1 requires the MSSVs to be operable in MODES 1, 2, and 3. Inoperability of the MSSVs, as determined during the testing, required entry into Condition A of LCO 3.7.1. The required action for LCO 3.7.1, Condition A, required that thermal power be reduced to < 51% rated thermal power within 4 hours. No power reductions were required during the testing, because the valves were restored to operable status prior to the required completion time. Therefore, the reportability assessment of this condition is conservatively based on the guidance of NUREG-1022, Revision 2, which indicates that multiple test failures of this type represent a condition that, if known, would have required the reactor power level to be reduced and, hence, was a condition prohibited by the Technical Specifications.

An evaluation of the safety consequences of the condition was completed. The safety consequences were determined to be minimal, and the overpressure protection of the steam generators, main steam lines, and reactor coolant system [AB] would not be exceeded due to the higher lift pressures. Therefore, the failure of these four main steam safety valves did not trigger any other reporting criteria.

The corrective actions, as described in the corrective actions section of this report, were completed during Refueling Outage (RO) 21, which started on October 12, 2002, and ended on November 13, 2002. Work was performed on the entire set of 12 MSSVs during RO-21. The work scope required post-maintenance testing on seven of the 12 valves (SV1-1A, 1B, 1C, 2B, 2C, 3A, and 4C).

II. CAUSE OF EVENT

The root cause of the out of tolerance lift pressures for these four main steam safety valves was attributed to mechanical component failure/degradation due to slight binding of the spindle on the guide bearing. Wear was observed on the spindle that may have contributed to the interference between the guide bearing and spindle. It was judged that binding of the spindle on the thrust bearing and spring-drift (i.e., minor change in the spring characteristics based on environmental conditions) were likely contributing causes. (A valve diagram is provided as Figure 1).

III. ANALYSIS OF EVENT

Technical Specifications Table 3.7.1-2 lists the main steam safety valve lift setting requirements as follows:

VALVE NUMBER			LIFT SETTING (psig +/- 3%)
STEAM GENERATOR			
A	B	C	
SV1-1A	SV1-1B	SV1-1C	1085
SV1-2A	SV1-2B	SV1-2C	1110
SV1-3A	SV1-3B	SV1-3C	1125
SV1-4A	SV1-4B	SV1-4C	1140

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

The four valves that failed to meet the +/- 3% acceptance criteria were SV1-1A, SV1-2B, SV1-1C, and SV1-2C. The results of the "as-found" test for each valve is provided in the following table:

VALVE NUMBER	AS-FOUND LIFT PRESSURE (PSIG)	PERCENT EXCEEDANCE	STEAM GENERATOR
SV1-1A	1119	3.1%	A
SV1-2B	1156	4.1%	B
SV1-1C	1126	3.8%	C
SV1-2C	1187	6.9%	C

As stated in the bases for LCO 3.7.1, the primary purpose of the MSSVs is to provide overpressure protection for the secondary system. The MSSVs also provide protection against overpressurizing the reactor coolant pressure boundary (RCPB) by providing a heat sink for the removal of energy from the Reactor Coolant System (RCS) if the preferred heat sink, provided by the Condenser and Circulating Water System, is not available.

Four MSSVs are located on each main steam header, outside containment, upstream of the main steam isolation valves, as described in the Updated Final Safety Analysis Report (UFSAR), Section 10.3.2. The MSSVs must have sufficient capacity to limit the secondary system pressure to </= 110% of the steam generator design pressure in order to meet the requirements of the ASME Code, Section III. The MSSV design includes staggered setpoints, according to Technical Specifications Table 3.7.1-2 in the accompanying LCO, so that only the needed valves will actuate. Staggered setpoints reduce the potential for valve chattering that could result from steam pressure insufficient to fully open all valves following a turbine or reactor trip.

The design basis for the MSSVs comes from ASME Code, Section III, and their purpose is to limit the secondary system pressure to </= 110% of design pressure for any anticipated operational occurrence (AOO) or accident considered in the design basis accident (DBA) and transient analysis.

The events that challenge the relieving capacity of the MSSVs, and thus RCS pressure, are those characterized as decreased heat removal events, which are presented in the UFSAR, Section 15.2. Of these, the loss of external electrical load is the limiting AOO. This event also terminates normal feedwater flow to the steam generators.

An evaluation of the effect of the as-found lift pressures for the MSSVs was performed. This evaluation concluded that the as-found lift pressures for the identified MSSVs would not cause any safety analysis limits to be exceeded.

IV. CORRECTIVE ACTIONS

The following table summarizes the work performed on the MSSVs during RO-21:

VALVE NUMBER	WORK COMPLETED DURING RO-21
SV1-1A	The valve was disassembled, inspected, the spindle was replaced, and the valve seats were lapped.
SV1-3A	The valve was disassembled, inspected, the spindle was replaced, and the valve seats were lapped.
SV1-1B	The valve was disassembled, inspected, the spindle was replaced, the guide bearing was machined, and the valve seats were lapped.
SV1-2B	The valve was disassembled, inspected, the spindle was replaced, the guide bearing was machined, and the valve seats were lapped.
SV1-1C	The valve was disassembled, inspected, the spindle was replaced, and the valve seats were lapped.
SV1-2C	The valve was disassembled, inspected, the guide bearing was machined, the spindle was polished, and the valve seats were lapped.
SV1-4C	The valve was disassembled, inspected, the spindle was replaced, and the valve seats were lapped.
Entire Set of 12 Valves	Removed spindle nut and cotter pin. This change is considered an enhancement and is not expected to significantly affect valve performance.

The work performed on the valves, in conjunction with the testing performed prior to and during RO-21, provides assurance that the valves are operable. No additional corrective actions are required. The test failure of four MSSVs has caused the maintenance rule (10 CFR 50.65) reliability criterion for the MSSVs to be exceeded, which in turn requires monitoring the MSSVs in accordance with 10 CFR 50.65(a)(1). The system will be returned to 10 CFR 50.65(a)(2) status in accordance with the guidance contained in the Progress Energy Nuclear Generation Group procedure, ADM-NGGC-0101, "Maintenance Rule Program," which requires corrective actions, goal setting, and performance monitoring.

V. ADDITIONAL INFORMATION

A. Failed Component Information:

The MSSVs [System Code: SB and Component Code: RV] are style HC-65W, Type C, manufactured by the Crosby Valve and Gage Company. The SV1-1 and 2 valves are type 6Q8, with a mean seat area of 15.473 square inches. The SV1-3 and 4 valves are type 6R10, with a mean seat area of 22.438 square inches.

B. Previous Similar Events:

On April 5, 2001, SV1-1C failed the as-found lift pressure test acceptance criteria of +/- 3% The valve lift pressure was measured as 1211 psig. The maximum acceptable as-found lift pressure for this valve is 1108 psig. The valve spindle was replaced and the guide bearing clearance was increased to reduce the possibility of valve binding. The valve performance, although found to be out of tolerance during testing on October 9, 2002, was substantially improved as demonstrated by the as-found lift pressure test data (i.e., in 2001 the as-found lift pressure was 1211 psig and in 2002 the as-found lift pressure was 1126 psig).

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Figure 1
Main Steam Safety Valve Diagram