BALTIMORE GAS AND ELECTRIC

CALVERT CLIFFS NUCLEAR POWER PLANT
1650 CALVERT CLIFFS PARKWAY • LUSBY, MARYLAND 20657-4702

CHARLES H. CRUSE
PLANT GENERAL MANAGER
CALVERT CLIFFS

March 18, 1993

U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

ATTENTION: Document Control Desk

SUBJECT: Calvert Cliffs Nuclear Power Plant
Unit No. 2; Docket No. 50-318; License No. DPR 69
Licensee Event Report 93-001
Pressurizer Code Safety Valve High As-Found Setpoints

Gentlemen:

The attached report is being sent to you as required under 10 CFR 50.73 guidelines. Should you have any questions regarding this report, we will be pleased to discuss them with you.

Very truly yours,

Charles H. Cruse

CHC/WDM/bjd
Attachment

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(5-92)

U. S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES: 5/31/952

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)
Calvert Cliffs, Unit 2	05000 318	1 OF 06

TITLE (4)
Pressurizer Code Safety Valve High As-Found Setpoints

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 NUMBERS(S)
										05000
02	20	93	93	-- 001 --	00	03	18	93		05000

OPERATING MODE (9): 3

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 (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 (include Area Code)
Wayne D. Maki, Compliance Engineer	410-260-3651

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

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS
D	AB	R V	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-space typewritten lines) (16)

On February 20, 1993, Calvert Cliffs Nuclear Power Plant (CCNPP) Unit 2 was found to be in a condition outside its design basis when both Pressurizer Code Safety Valves were found with lift setpoints higher than those allowed by the Technical Specifications and above those used in the plant's safety analyses. At the time of discovery, CCNPP Unit 2 was in Hot Standby (Mode 3) with reactor coolant temperature and pressure at approximately 530 degrees Fahrenheit and 1990 PSIG. The discovery was made during surveillance testing associated with a normal shutdown preceding a scheduled refueling outage. Both valves had last been tested in March 1989.

Imprecision associated with the test methodology was the primary causal factor. One of the valves will be inspected by the vendor for mechanical degradation, but it is thought unlikely that this was a factor.

Immediate corrective action was taken to reset the valves to within specification. Refinements to the test procedure are being evaluated. Both valves will have their setpoints verified using new testing methodology prior to power operation of Unit 2 following the current refueling outage.

I. DESCRIPTION OF EVENTS

On February 20, 1993 Calvert Cliffs Nuclear Power Plant (CCNPP) Unit 2 was found to be in a condition outside the design basis of the plant when both Pressurizer Code Safety Valves (PSVs) were found with lift setpoints above those required by Technical Specification (TS) 3.4.2.1 and above those used in the plant's safety analyses. The discovery was made during routine surveillance testing conducted during a normal shutdown at the beginning of a scheduled refueling outage for Unit 2. At the time, the plant was in hot standby (Mode 3) with reactor coolant system temperature and pressure at approximately 530 degrees and 1990 PSIG. The valves were tested in place using the hydroset technique.

CCNPP utilizes two PSVs, 2-RC-200-RV and 2-RC-201-RV. TS 3.4.2.1 requires both PSVs to be operable in modes 1,2, and 3 with lift settings in the following pressure ranges:

Valve	Required Setpoint	Setpoint Range
2-RC-200-RV	2500 (+/- 1 percent) PSIA	2475-2525 PSIA
2-RC-201-RV	2565 (+/- 1 percent) PSIA	2540-2590 PSIA

The TS ACTION statement requires that with one PSV inoperable in Modes 1, 2, or 3, "either restore the inoperable valve to OPERABLE status within 15 minutes or be in HOT SHUTDOWN within 12 hours."

The lift setpoint of 2-RC-200-RV was found to be 2593 PSIA at 12:00 noon, 68 PSI higher than the maximum specified value of 2525 PSIA. The valve was declared inoperable and immediate action was taken to reset the safety valve to within specification. The valve was satisfactorily adjusted and declared OPERABLE at 5:00 p.m. The surveillance was then performed on 2-RC-201-RV and its setpoint was found to be 2625 PSIA, 35 PSI more than the maximum allowed. The valve was declared inoperable at 10:15 p.m. and immediately readjusted to within specifications. The valve was declared OPERABLE at 10:55 p.m.

After the condition was discovered and the PSVs reset, CCNPP continued the ongoing controlled shutdown in order to perform maintenance and refueling unrelated to this event. No failures of other plant systems or components resulted from the event. The valves had both been last tested in March 1989.

II. CAUSE OF EVENTS

A formal root cause analysis was conducted to determine the factors leading to the out of specification setpoints. One of the valves, 2-RC-200-RV was previously scheduled to be removed for overhaul. The valve will be disassembled and inspected by the vendor to determine if mechanical degradation was a factor. Final determination of the root cause must await the results of this inspection. However, we consider a mechanical mode of failure less likely as our experience indicates that most mechanical wear results in a valve setpoint reduction. The valve vendor agrees with this observation.

The analysis identified imprecision associated with the test methodology to be the primary cause. The surveillance procedure, STP-M-2-2, "Pressurizer Safety Valves Setpoint Adjustment," requires the technician to record the point where the valve "simmers" as the point of lift. This term is not further defined and is subject to interpretation. The technicians performing the test in this case waited until they heard a fairly loud noise before recording the lift point, since ambient noise levels were quite high inside containment where the valves are located. Based upon observation there is a 30-60 PSI difference between when the valve first makes a detectable noise and when it makes a clearly audible rumble. Additionally, a digital pressure gauge was used on the test device. In past performances of this test, an analog gauge was used. By watching for a change of the rate of pressure increase on the analog gauge, the technician was given an additional clue as to when the valve was first starting to lift. Without this additional clue, the technicians had to wait longer before they were certain that the valve was lifting.

Another contributing factor was a recent change in the definition of the "as-found" setpoint. Previously, the very first lift (characteristically 10-30 psi higher than subsequent lifts) was disregarded per the vendor's recommendation. This was based on the importance of obtaining a repeatable result on which to base adjustments. In order to obtain the most conservative value for the as-found setpoint, however, the procedure now requires the actual first lift to be used for determining Technical Specification compliance.

III. ANALYSIS OF EVENT

This event is reportable under 10 CFR 50.73(a)(2)(ii)(B) because the maximum lift settings allowed by the Technical Specifications are used in the plant's UFSAR Chapter 14 safety analyses, and exceeding these settings therefore placed the plant in a condition outside its design basis.

Two PSVs are located on the pressurizer to provide overpressure protection for the reactor coolant system. They are totally enclosed, back-pressure compensated, spring-loaded safety valves which meet ASME Code requirements. The safety valves were designed to pass sufficient steam to limit the primary system pressure to 110 percent of design (2750 PSIA) following a complete loss of turbine generator load without simultaneous reactor trip while operating at 100 percent rated thermal power (2700 MWt). The reactor is assumed to trip on a high reactor coolant system (RCS) pressure signal. To determine the maximum steam flow, the only other pressure relieving system assumed operational is the Main Steam Safety Valves. Conservative values are assumed for all system parameters, delay times, and core moderator coefficients. The loss of load event is described in the UFSAR Section 14.5.

The PSVs are also credited in the mitigation of reactor coolant system overpressurization during a worst case break of the main feedwater system piping and during a loss of feedwater event. The more limiting feed line break (FLB) analysis is described in the UFSAR Section 14.26.

The peak reactor coolant system pressures derived from the above analyses are approximately 2700 PSIA for the Loss of Load event and 2749 PSIA for the FLB event. To derive a new peak pressure from the as-found setpoints, a conservative method would be to take the maximum error and add it to the above values. This results in a new peak pressure of 2768 PSIA and 2817 PSIA. The maximum theoretical reactor coolant pressure observed would therefore be 2817 PSIA for the feed line break, or roughly 2.5 percent in excess of the pressure upset limit. Exceeding this limit by such a small amount would not threaten the integrity of the reactor coolant system, since the system was designed with a factor of safety of three per Section III of the ASME Boiler and Pressure Vessel Code. A large factor of safety would still exist even with the increased peak pressure.

However, it can also be reasonably expected that several mitigating factors would act to prevent the RCS pressure from exceeding the pressure upset limit. Several conservatisms are included in these analyses. No credit is taken for the operation of the Power Operated Relief Valves, both of which were operable in this case. No credit is taken for operation of the steam dump or bypass valves, operation of which would greatly reduce the extent of the pressure transient. To maximize the peak pressure obtained in the analyses, the only reactor trip credited is the high pressurizer pressure trip. For the loss of load event, the expected reactor trip following the turbine trip would mitigate the pressure rise. For the Feed Line Break event, a high containment pressure or low steam generator water level trip would do the same. Credit is also not

taken for operation of the pressurizer level and pressure control systems. It can reasonably be assumed that all of the above systems would have functioned as designed to mitigate the consequences of a loss of load or feed line break event. The peak pressure seen in a credible incident would therefore have been well below the pressure upset limit.

Though the safety margin was reduced a small amount with the Pressurizer Safety Valves in the as-found condition, there clearly was no significant threat to the reactor coolant system's integrity or the health and safety of the public.

IV. CORRECTIVE ACTIONS

The immediate corrective action for the event was to reset the valves to within specification. Both valves were reset within eleven hours of the first valve being found out of tolerance. A generic operability review of the Unit 1 pressurizer code safety valves was conducted. Differences between the conduct of the last surveillance on these valves and the event reported here provide a high degree of confidence that the Unit 1 PSV setpoints are within specification.

Improvements in test methods being evaluated include incorporation of a less subjective definition of the "simmer" point, and using the analog vs. digital gauge for hydroset pressure measurement.

Both valves will have their setpoints verified using the more precise testing method prior to power operation of the unit following its current refueling outage.

V. ADDITIONAL INFORMATION

A. Identification of components referred to in this LER

Component	IEEE 803 EIIS Function	IEEE 805 System ID
Pressurizer Safety Relief Valve	RV	AB
Reactor Coolant System	N/A	AB
Main Steam Safety Valves	RV	SB
Turbine Generator	TG	TB
Main Feedwater System	N/A	SJ
Check Valve	ISV	SJ
Power Operated Relief Valve	RV	AB

B. Previous Similar Events

There have been two previous similar events involving out of specification PSVs at CCNPP Unit 1 (LERs 87-06, and 88-11) and one previous similar event at CCNPP Unit 2 (LER 85-10). Similar events are common in the industry.

No other component failures resulted from this event. The Pressurizer Code Safety Valves 2-RC-200-RV and 2-RC-201-RV are Dresser Industries type 31739A Maxiflow Safety Valves.