NRC FORM 366 (7-77)

U. S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT

CONTROL BLOCK: (1) (PLEASE PRINT OR TYPE ALL REQUIRED INFORMATION)

01	LICENSEE CODE (2)	NJSGS1
01	LICENSE NUMBER (3)	00-00000-00
01	LICENSE TYPE (4)	41111
01	CAT (5)	
01 (CON'T)	REPORT SOURCE (6)	L
01	DOCKET NUMBER (7)	05000272
01	EVENT DATE (8)	031183
01	REPORT DATE (9)	071383

EVENT DESCRIPTION AND PROBABLE CONSEQUENCES (10)

02 Pressurizer Code Safety Valves 1PR3, 1PR4, and 1PR5 were tested for lift set pressure
03 and seat leakage by Wyle Laboratories during the period of November 2-13, 1982. All
04 valves lifted in excess of the 2485 ± 1% pressure range specified in Technical
05 Specification 3.4.2.2. Also, all valves exhibited heavy seat leakage. The test lift
06 pressures were: 1PR3 - 2564 psig (54 psig over), 1PR4 - 2532 psig (22 psig over),
07 1PR5 - 2546 psig (36 psig over). Engineering evaluation revealed that the setpoint
08 deviations did not present a safety concern.

09	SYSTEM CODE (11)	CAUSE CODE (12)	CAUSE SUBCODE (13)	COMPONENT CODE (14)	COMP. SUBCODE (15)	VALVE SUBCODE (16)
	CB	D	Z	VALVEX	J	B

(17) LER/RO REPORT NUMBER

EVENT YEAR		SEQUENTIAL REPORT NO.		OCCURRENCE CODE	REPORT TYPE		REVISION NO.
83	—	008	/	03	X	—	1

ACTION TAKEN (18)	FUTURE ACTION (19)	EFFECT ON PLANT (20)	SHUTDOWN METHOD (21)	HOURS (22)	ATTACHMENT SUBMITTED (23)	NPRD-4 FORM SUB. (24)	PRIME COMP. SUPPLIER (25)	COMPONENT MANUFACTURER (26)
D	G	Z	Z	0000	Y	Y	N	C710

CAUSE DESCRIPTION AND CORRECTIVE ACTIONS (27)

10 All three valves were repaired and retested satisfactorily, then reinstalled. During
11 subsequent testing of Salem Unit 2 valves it was revealed that the setpoint deviation
12 and leakage were apparently due to the testing methods utilized. The valves will be
13 tested again during the next refueling; support will be provided to improve testing
14 methods.

15	FACILITY STATUS (28)	H
15	% POWER (29)	000
15	OTHER STATUS (30)	N/A
15	METHOD OF DISCOVERY (31)	C
15	DISCOVERY DESCRIPTION (32)	Pressure Lift Test
16	ACTIVITY RELEASED (33)	Z
16	CONTENT OF RELEASE (34)	Z
16	AMOUNT OF ACTIVITY (35)	N/A
16	LOCATION OF RELEASE (36)	N/A
17	PERSONNEL EXPOSURES NUMBER (37)	000
17	TYPE (38)	Z
17	DESCRIPTION (39)	N/A
18	PERSONNEL INJURIES NUMBER (40)	000
18	DESCRIPTION (41)	N/A
19	LOSS OF OR DAMAGE TO FACILITY TYPE (42)	Z
19	DESCRIPTION (43)	N/A
20	PUBLICITY ISSUED (44)	N
20	DESCRIPTION (45)	N/A
20	NRC USE ONLY	

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PDR ADOCK 05000272
S PDR

IE22

NAME OF PREPARER R. Frahm PHONE: (609) 935-6000 Ext. 4309

GPO 917-926

PSE&G

Public Service Electric and Gas Company P.O. Box E Hancocks Bridge, New Jersey 08038

Salem Generating Station

July 27, 1983

Dr. Thomas E. Murley
Regional Administrator
USNRC
Region 1
631 Park Avenue
King of Prussia, Pennsylvania 19406

Dear Dr. Murley:

LICENSE NO. DPR-70
DOCKET NO. 50-272
REPORTABLE OCCURRENCE 83-008/03X-1
SUPPLEMENTAL REPORT

Pursuant to the requirements of Salem Generating Station Unit No. 1 Technical Specifications, Section 6.9.1.9.b, we are submitting supplemental Licensee Event Report for Reportable Occurrence 8.-008/03X-1.

Sincerely yours,

J. M. Zupko, Jr.

J. M. Zupko, Jr.
General Manager -
Salem Operations

RF:kl

CC: Distribution

IE22

Report Number: 83-008/03X-1

Report Date: 07-13-83

Occurrence Date: 03-11-83

Facility: Salem Generating Station, Unit 1
Public Service Electric & Gas Company
Hancocks Bridge, New Jersey 08038

IDENTIFICATION OF OCCURRENCE:

Pressurizer Code Safety Valves - Inoperable

This report was initiated by Incident Report 83-052.

CONDITIONS PRIOR TO OCCURRENCE:

Mode 5 - Rx Power 0% - Unit Load 0 MWe.

DESCRIPTION OF OCCURRENCE:

Pressurizer Code Safety Valves 1PR3, 1PR4, and 1PR5 were tested for lift set pressure and seat leakage by Wyle Laboratories during the period of November 2-13, 1982. All valves lifted in excess of the 2485 psig $\pm$ 1% pressure range specified in Technical Specification 3.4.2.2. All valves displayed heavy seat leakage. The actual lift pressures were: 1PR3 - 2564 psig (54 psig over), 1PR4 - 2532 psig (22 psig over), 1PR5 - 2546 psig (36 psig over). These valves had been tested, repaired, and retested during the first refueling outage in 1979. They were within specification when reinstalled. No Reactor Coolant System (RCS) pressure transients occurred during previous power operation which resulted in actuation of the safety valves; the integrity of the RCS and redundant fission product barriers was maintained.

DESIGNATION OF APPARENT CAUSE OF OCCURRENCE:

Subsequent investigation revealed that leakage observed through the valves could be attributed partly to the difference in test versus actual operating conditions. During testing, the valve was directly placed above the pressurizing chamber and was subjected to temperatures close to 500°F. In the plant, a long water loop seal precedes each valve, and each valve body inlet temperature is less than 200°F. The valve internals are of a type that provides leak tightness during plant operation and are not designed to be tight against high temperature steam. This was substantiated by nitrogen tests where a Salem Unit 2 valve tested leak tight (while during tests at the same setpoint with steam the valve leaked). A steam test followed the nitrogen test, with the body temperature limited to 200°F. The valve was successfully lift tested with no subsequent seat leakage. It was therefore tentatively assumed that the setpoint variations were related to the elevated test temperatures utilized.

ANALYSIS OF OCCURRENCE:

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its safety limit of 2735 psig. Each safety valve is designed to relieve 420,000 pounds per hour of saturated steam at the valve setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are operable, an operating RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization. In addition, the Overpressure Protection System provides a diverse means of protection against RCS overpressurization at low temperatures.

During operation, all pressurizer code safety valves must be operable to prevent the RCS from being pressurized above its safety limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss of load assuming no reactor trip until the first Reactor Protective System trip setpoint is reached (i.e., no credit is taken for a direct reactor trip on the loss of load) and also assuming no operation of the power operated valves or steam dump valves.

Evaluation of the potential impact of the possible safety valve setpoint variation on plant performance during the analyzed transient was performed. The evaluation S-C-R200-NSE-193 states:

> Valve opening pressures in excess of assumed were generically evaluated by Westinghouse. The limiting overpressurization transient evaluated for a four-loop unit is the loss of external load and/or turbine trip without immediate reactor trip. For this event, safety valve functioning is not required if the reactor trips on high pressurizer pressure. If the reactor does not trip until the second protection grade trip (overtemperature delta T), a valve opening delay time of approximately two seconds would still provide acceptable overpressure protection for the reactor coolant system and all components would be exposed to a pressure within 110 percent of the system design pressure.
>
> Although set pressure deviation varied from 22 psig to 54 psig (0.8 to 2.2%) above the allowable band, effective relief would have started at the 22 psig shift above allowable (2,485 + 1%, i.e., 2,510 psig). A two second delay in safety valve openings as analyzed by Westinghouse would be the limiting case as opposed to 22 psig to 54 psig, setpoint deviation at Salem Unit 1 based on the rate of pressure rise from Salem FSAR for the accident condition analyzed.

ANALYSIS OF OCCURRENCE: (continued)

Based on the generic evaluation as above, it is judged that upward deviation of Salem 1 pressurizer safety valve setpoint during operation did not present a safety concern.

Apart from the qualitative difference between the test and operating condition, a leaky valve does not pose any safety concern, as long as the Unit Technical Specification governing the allowable RCS leakage is not violated.

As noted, no pressure transient was involved and the integrity of multiple fission product barriers was maintained. Finally, the problems apparently involved testing methods and not an actual variation in valve setpoints. The occurrence therefore constituted no undue risk to the health or safety of the public. Due to the potential for operation in a degraded mode permitted by a Limiting Condition for Operation the event is reportable in accordance with Technical Specification 6.9.1.9b.

CORRECTIVE ACTION:

The valve manufacturer, in conjunction with Wyle engineers, refurbished the valves by ultrasonic cleaning, lapping the seating surfaces and reestablishing the ring positions. Subsequent retesting indicated the opening pressures to be within the allowable tolerance and no leakage was observed.

The refurbished and retested valves have shown acceptable performance. The valves will be tested again during the next refueling. At that time support will be provided to insure that testing performed more closely models actual valve operating conditions.

FAILURE DATE:

Crosby Valve and Gage Co.
Pressurizer Safety Valve
Part No. HB-86-BP

Prepared by R. Frahm

J. M. Zupko
General Manager -
Salem Operations

SORC Meeting No. 83-094