PSEG

Public Service Electric and Gas Company P.O. Box 236 Hancocks Bridge, New Jersey 08038-0236

Nuclear Business Unit

NOV 22 1995

LR-N95208

U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Gentlemen:

LICENSEE EVENT REPORT 272/95-026-00
SALEM GENERATING STATION - UNIT 1
FACILITY OPERATING LICENSE NO. DPR-70
DOCKET NO. 50-272

This Licensee Event Report entitled "Main Steam Safety Valves Failed Lift Set Test" is being submitted pursuant to the requirements of the Code of Federal Regulations 10CFR50.73 (a)(2)(i).

Sincerely,

Clay C. Warren
General Manager -
Salem Operations

SORC Mtg. 95-140
Attachment

JHA/tcp

C Distribution
LER File 3.7

IE22
1/1

Attachment A

PSE&G Commitments for LER 272/95-026

The following item represents PSE&G commitments made to the Nuclear Regulatory Commission related to LER 272/95-026-00. The commitment is as follows:

A review of past maintenance and performance histories for all MMSVs (Units 1 and 2) for generic implications. This review will be completed by December 15, 1995.

Appropriate enhancements will be made to the safety valve program based on the results of the root cause determination. These enhancements will be complete by June 1, 1996.

Additional corrective actions identified based on the results of the on-going cause investigation will be reported in a supplement to this LER. The expected submission date for the supplement is March 1, 1996.

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

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

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS MANDATORY INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F33), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT

LICENSEE EVENT REPORT (LER)

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

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
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TITLE (4)

Main Steam Safety Valves Failed Lift Set Test

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	-- 026 --	000	11	22	95		05000
									FACILITY NAME	DOCKET NUMBER
										05000

OPERATING MODE (9)	*
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)	X	50.73(a)(2)(i)
	20.2203(a)(1)		20.2203(a)(3)(i)		50.73(a)(2)(ii)
	20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)
	20.2203(a)(2)(ii)		20.2203(a)(4)		50.73(a)(2)(iv)
	20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)
	20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)

	50.73(a)(2)(viii)
	50.73(a)(2)(x)
	73.71
	OTHER
	Specify in Abstract below or in NRC Form 366A

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Lee Parris, IST Group Supervisor	609 - 339 - 2024

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

SUPPLEMENTAL REPORT EXPECTED (14)			EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
X	YES (If yes, complete EXPECTED SUBMISSION DATE).	NO		03	01	96

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

Between October 23 and November 15, 1995, during scheduled surveillance testing, it was discovered that nine out of twenty Main Steam Safety Valves (MSSV) exceeded the allowable lift set pressure tolerance specified in Technical Specification Table 4.7-1. Three of the nine valves lifted above the allowable setpoint while six lifted below the allowable setpoint. Testing was being performed as required by Technical Specification 4.0.5.

The cause of these occurrences is under investigation and will be reported in a supplement to this LER. Additional corrective actions will be reported in the supplement as well.

This event is reportable per 10CFR50.73(a)(2)(i)(B), operation prohibited by the plant's Technical Specifications.

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

PLANT AND SYSTEM IDENTIFICATION

Westinghouse - Pressurized Water Reactor

Main Steam / Safety Valves {SB/RV}*

* Energy Industry Identification System (EIIS) codes and component function identifier codes appear in the text as {SS/CCC}.

IDENTIFICATION OF OCCURRENCE

Event Date: October 23, 1995 (first occurrence)

Report Date: November 22, 1995

CONDITIONS PRIOR TO OCCURRENCE

Defueled - 0 % reactor power

DESCRIPTION OF OCCURRENCE

Between October 23 and November 15, 1995 nine of the twenty Main Steam Safety Valves (MSSV) {SB/RV} tested to satisfy Technical Specification 4.0.5 requirements failed to meet the allowable lift set pressure tolerance. These valves are Category C valves per ASME Section XI and as such, are subjected to a lift set test on a five year frequency. Technical Specification Table 4.7-1 specifies the lift setpoint pressure and requires that the valve lift set pressure be within plus or minus 1 percent of the valve lift setpoint pressure.

The test results for valves failing to meet the required lift set tolerance are presented in the following table:

Valve Number	Setpoint	Test Result	Deviation %
12MS11	1125 psig	1159 psig	plus 3.02
12MS13	1110	1047	minus 5.68
13MS13	1110	1124	plus 1.26
13MS14	1100	1128	plus 2.55
14MS11	1125	1090	minus 3.11
14MS12	1120	1084	minus 3.21
14MS13	1110	1086	minus 2.16
14MS14	1100	1052	minus 4.36
14MS15	1070	1059	minus 1.03

Testing of all twenty MSSVs was completed on November 15, 1995.

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

APPARENT CAUSE OF OCCURRENCE

The investigation to determine the cause of these occurrences is on-going. Areas currently under investigation include maintenance history and test conditions. The cause of these occurrences will be reported in a supplement to this LER following completion of the investigation.

PRIOR SIMILAR OCCURRENCES

LER 272/89-003-00 reported a reactor trip due to steam flow/feed flow mismatch. During the post trip recovery, the 23MS15 MSSV lifted. The valve appeared to lift at 1030 psig, below the required setpoint of 1070 psig. The premature lifting of the valve was attributed to setpoint drift.

LER 311/89-008-00 reported a reactor trip due to equipment failure. During the event, the 24MS15 MSSV lifted twice. Investigation revealed that the lift set pressure was 901 psig and a proper setpoint of 1070 psig could not be achieved. The valve was replaced.

LER 272/89-024-00 reported a safety injection and reactor trip due to inadequate procedures. During the event, the 13MS15 MSSV lifted twice due to density wave induced pressure spikes. Following the event, all MSSVs were tested and ten were found to have lift set pressures below the allowable value. The cause was suspected to be that the valve lift pressures were set under conditions that did not adequately account for the ambient temperature that the installed valves would experience. All out of tolerance valves were reset.

In addition, as reported in LER 272/83-008/03X-1 and 272/95-005-001, there have also been previous occurrences of pressurizer relief valves {AB/RV} failing to meet the Technical Specification lift set allowable value. The cause for the 1983 occurrences was attributed to testing methodology. The cause for the 1995 occurrences was attributed to instrument error, shipping and handling effects, and applied loads from discharge piping.

SAFETY SIGNIFICANCE

The Salem licensing basis UFSAR Chapter 15 accident analyses has been recently re-analyzed in support of a Fuel Upgrade / Margin Recovery Program (FUMRP). The analyses performed for this FUMRP have not been formally incorporated into the licensing basis due to the necessary NRC reviews. However, the FUMRP analyses bound the current plant configuration and assume a plus or minus 3 percent MSSV set pressure drift, and therefore are being used to assess safety significance of MSSV set pressure drift reported here.

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

SAFETY SIGNIFICANCE (cont'd)
All of the Chapter 15 accidents analyzed, except the Steam Generator Tube Rupture, are adversely affected if the MSSV set pressure increases and have been analyzed for the FUMRP to bound a plus 3 percent set pressure drift. The Steam Generator Tube Rupture accident analysis is adversely affected by a lower MSSV set pressure and bounds a minus 3 percent MSSV set pressure drift.

The deviations in the high pressure direction in the test data reported here are at or below that assumed in the FUMRP (plus 3 percent). The highest MSSV set pressure test result achieved lift at 1159 psig. This is listed as 3.02 percent out of tolerance. However, the FUMRP analyses referenced earlier assumed the highest MSSV (1125 psig) to lift at 1125 X 1.03 or 1159 psig (rounded from 1158.75). Thus, those high deviations are bounded by the FUMRP analyses. The MSSVs which drifted in the low pressure direction, drifted no lower than 1047 psig. This is a higher pressure than the lowest MSSV set pressure (1070 psig) minus 3 percent. Thus, while the MSSVs drifted lower than 3 percent from their individual set pressures, the drift was no lower than 3 percent of the lowest set MSSV (1070 psig). The FUMRP Steam Generator Tube Rupture analysis models only the lowest MSSV minus 3 percent drift, so that the MSSV set pressure drift reported here in the low pressure direction is bounded by the FUMRP analyses. Therefore, due to to the FUMRP analyses bounding the test results, the conclusions of the current UFSAR Chapter 15 licensing basis analyses remain valid.

CORRECTIVE ACTIONS

Corrective actions will include the following:

A review of past maintenance and performance histories for all MMSVs (Units 1 and 2) for generic implications. This review will be completed by December 15, 1995.

Appropriate enhancements will be made to the safety valve program based on the results of the root cause determination. These enhancements will be complete by June 1, 1996.

Additional corrective actions identified based on the results of the on-going cause investigation will be reported in a supplement to this LER.