NRC Form 366
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/85

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Salem Generating Station - Unit 1	0 5 0 0 0 2 7 2	1 OF 0 3

TITLE (4)
Reactor Trip From 60% - Low Low Level No. 13 Steam Generator

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 1	2 1	8 4	8 4	— 0 0 4	— 0 0	0 2	1 7	8 4		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9)	1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)			
POWER LEVEL (10)	0 6 0	20.402(b)	20.405(c)	X 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)	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
J. L. Rupp	6 0 9 — 3 3 9 - 4 3 0 9

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	SJ	P D C V	M 1 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-spaced typewritten lines) (16)

On January 21, 1984, a reactor trip occurred due to a low-low water level in No. 13 Steam Generator. The initiating event was the closure of the inlet valve to a Low Pressure Feedwater Heater string, due to a high level spike in a feedwater heater. The bypass valve around the heaters (1CN47) failed to open automatically, and would not respond to remote-manual control signals. This caused No. 11 Steam Generator Feedwater Pump to trip on low suction pressure, which led to the low level condition in the steam generator. Water level decreased to the low-low level setpoint, which caused the reactor trip. 1CN47 was found to be intermittently binding when in the shut position. A similar occurrence on January 7, 1984 was documented in LER 84-002-00. Extensive testing, at that time, did not reveal the intermittent binding problem. The valve has been adjusted to limit the travel, which will prevent the valve from binding when in the shut position. The purpose of the reactor trip is to prevent operation with the steam generator water level below the minimum volume required for adequate heat removal. The Reactor Protection System functioned as designed, and the heat sink was maintained. The event involved no undue risk to the health or safety of the public. Due to the automatic operation of the Reactor Protection System, the event is reportable in accordance with 10CFR50.73(a)(2)(iv).

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PLANT AND SYSTEM IDENTIFICATION:

Westinghouse - Pressurized Water Reactor

Energy Industry Identification System (EIIS) codes are identified in the text as [XX].

IDENTIFICATION OF OCCURRENCE:

Reactor Protection System [JC] - Reactor Trip From 60% - Low Low Level No. 13 Steam Generator - (Rx Trip #84-04)

Event Date: 01/21/84

Report Date: 02/17/84

This report was initiated by Incident Report No. 84-014

CONDITIONS PRIOR TO OCCURRENCE:

Mode 1 - Rx Power 060 % - Unit Load 0632 MWe

DESCRIPTION OF OCCURRENCE:

At 2240 hours, January 21, 1984, a reactor trip occurred due to a low water level condition in No. 13 Steam Generator. The initiating event was the closure of the inlet valve to 13A, 14A and 15A Low Pressure Feedwater Heater string (11CN27), due to a high level spike in 15A Feedwater Heater. The high level spike occurred while adjusting the controller for the Heater High Level Dump Valve (11HD65), to correct a heater high level condition. The bypass valve around the heaters (1CN47) failed to open automatically. An attempt was made to open 1CN47, using remote-manual control from the control room, but the valve failed to respond. This caused No. 11 Steam Generator Feedwater Pump [SJ] to trip on low suction pressure. The loss of the feedwater pump led to the low water level in No. 13 Steam Generator. Water level decreased to the low-low level setpoint, which caused the reactor trip.

APPARENT CAUSE OF OCCURRENCE:

1CN47 is designed to open automatically upon a 115 psi differential pressure across the heaters or, upon 265 psig suction pressure on the feed pumps. It is also designed to be operated from the control room. The valve failed to respond to automatic or manual control signals. On January 7, 1984, a similar occurrence was experienced; that occurrence was documented in LER 84-002-00. At that time, the differential pressure setpoint was found to be high. The valve was stroked and tested in all possible modes of operation. Remote-manual and automatic operation of the valve was satisfactory.

APPARENT CAUSE OF OCCURRENCE: (cont'd)

The valve operated smoothly, and no reason could be found for the valve not responding to the manual or the low suction pressure signals. Following this occurrence (January 21,1984), investigation revealed that the valve was experiencing an intermittent binding problem, while in the shut position.

ANALYSIS OF OCCURRENCE:

The purpose of the reactor trip, on low-low steam generator level, is to prevent operation with the steam generator water level below the minimum volume required for adequate heat removal; thereby preventing the loss of the reactor heat sink. The trip is actuated on two out of three low-low water level signals in any steam generator. The setpoint ensures that there is adequate inventory in the steam generators, at the time of the reactor trip, to allow for any possible starting delays of the Auxiliary Feedwater Pumps [BA]; thus preventing steam generator dry-out and the Reactor Coolant System [AB] thermal and hydraulic transients that would be associated with a loss of the heat sink. The Reactor Protection System [JC] functioned as designed, and the heat sink was maintained. The Reactor Coolant System has been designed to withstand the thermal and hydraulic effects of four-hundred (400) reactor trips from full power. This trip was well within the design limits of the system. This occurrence involved no undue risk to the health or safety of the public. Because of the automatic actuation of the Reactor Protection system, the event is reportable in accordance with the Code of Federal Regulations, 10CFR 50.73(a)(2)(iv).

CORRECTIVE ACTION:

As previously stated, 1CN47 was intermittently binding when in the shut position. The valve has been adjusted to limit the travel, which will prevent binding when in the shut position. It has been tested in all possible modes of operation. Remote-manual and automatic operation of the valve is satisfactory. Permanent repairs will be effected during the next available outage of sufficient duration.

General Manager-
Salem Operations

JLR:tms

SORC Mtg 84-018

PSE&G

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

Salem Generating Station

February 17, 1984

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

Dear Sir:

SALEM GENERATING STATION
LICENSE NO. DPR-70
DOCKET NO. 50-272
UNIT NO. 1
LICENSEE EVENT REPORT 84-004-00

This Licensee Event Report is being submitted pursuant to the requirements of 10CFR 50.73(a)(2)(iv). This report is required within thirty (30) days of discovery.

Sincerely yours,

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

JR:kll

CC: Distribution

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