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): SALEM GENERATING STATION UNIT - 1

DOCKET NUMBER (2): 0 5 0 0 0 2 7 2

PAGE (3): 1 OF 0 5

TITLE (4): REACTOR TRIPS FROM 91% and 93% DUE TO 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 5 0 0 0
1 1	0 6	8 4	8 4	0 2 5	0 0	1 2	0 6	8 4		0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 0 9 1

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)

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

SUPPLEMENTAL REPORT EXPECTED (14)

YES (If yes, complete EXPECTED SUBMISSION DATE) X NO

EXPECTED SUBMISSION DATE (15): MONTH | DAY | YEAR

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

On November 6, and November 11, 1984, malfunctions of the turbine Electro-Hydraulic Control System (EHCS) resulted in reactor trips. Rapid load rejection caused a sudden reduction of steam flow, a steam generator pressure increase and a "shrink" of the steam generator indicated level which resulted in reactor trips on No. 13 Steam Generator low-low water level signals. There was no loss of feedwater, and the "shrink" phenomena affected indicated level only. The Reactor Protection System (RPS) functioned as designed during both events. Following both reactor trips, extensive investigations and testing of the EHCS was performed by PSE&G with assistance from the vendor and consultants. Five printed circuit cards of high suspect were replaced, although none were positively identified as having initiated the events. A safety evaluation was performed which verified that the Unit could be safely operated under specific conditions set fourth in the evaluation. The Unit is presently operating under these requirements with continuous monitoring of the EHCS. No similar problems have been encountered to date. These events involved no undue risk to the health and safety of the public. Due to the automatic actuation of the RPS, the events are reportable in accordance with the code of Federal Regulations, 10CFR 50.73(a)(2)(iv).

IE22
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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 OCCURRENCES:

Reactor Trips From 91% and 93% Due to No. 13 Steam Generator Low-Low Water Level

Event Dates: 11/06/84

11/11/84

Report Date: 12/06/84

This report was initiated by Incident Reports **84-185** and **84-191**

CONDITIONS PRIOR TO OCCURRENCES:

11/06/84 - Mode 1 - Rx Power 091 % - Unit Load 1005 MWe

11/11/84 - Mode 1 - Rx Power 093 % - Unit Load 1010 MWe

DESCRIPTION OF OCCURRENCES:

At 0645 hours, November 6, 1984, during routine power operations, a reactor trip occurred due to No. 13 Steam Generator low-low water level. Approximately three to five (3-5) seconds prior to the trip, the "EH (Electro-Hydraulic) Speed/Load Channel Failure" and the "EH Protection System Trouble" alarms were received on the overhead annunciator in the control room. In addition, the steam generator "High Steam Flow" alarms were flashing on the RP-4 Panel. The Unit was stabalized in Mode 3. At 0659 hours, in accordance with the requirements of the Code of Federal Regulations, 10CFR 50.72(b)(2)(ii), the Nuclear Regulatory Commission was notified of the automatic actuation of the Reactor Protection System [JC].

This reactor trip was apparently initiated by a malfunction of the turbine Electro-Hydraulic Control (EHC) System, which resulted in a load rejection. This sudden reduction in steam flow resulted in a rapid increase of steam generator pressure and a "shrink" of the steam generator indicated level, which caused a reactor trip on low-low water level. Following the reactor trip, the condensate pumps were inadvertently removed from service. This resulted in saturation conditions and flashing in the Condensate System [SD].

DESCRIPTION OF OCCURRENCES: (cont'd)

The Condensate System was thoroughly inspected for damage. All inspections, leak tests and functional tests were satisfactory with the following exception. One pipe hanger was found to be damaged, and the hanger was replaced. Extensive investigations and testing of the EHC System was performed by PSE&G with assistance from the vendor and consultants. The system was thoroughly tested with the aid of a simulator; however, the results were inconclusive and the component responsible for the malfunction could not be positively identified. Various inputs of the EHC System were instrumented for a subsequent startup in an attempt to identify the portion of the system responsible for the malfunction, in the event that the malfunction repeated itself. In addition, after reviewing the SOER which documented similar problems encountered at another utility, one of the speed error printed circuit cards was replaced.

The Unit was returned to service at 0732 hours, November 9, 1984. Later that day, a partial load rejection was experienced. The EHC System was shifted to manual, and a complete load rejection and reactor trip was prevented. The instrumented system indicated a speed circuit fluctuation; as a result, the second speed error printed circuit card was replaced. The Unit then operated satisfactorily until 0109 hours, November 11, 1984, when the Unit again experienced a rapid load rejection. This load rejection resulted in a second reactor trip caused by No. 13 Steam Generator low-low water level signal. The conditions associated with this trip were very similar to those experienced on November 6, 1984. The Unit was stabalized in Mode 3. At 0111 hours, in accordance with the requirements of the Code of Federal Regulations, 10CFR 50.72(b)(2)(ii), the Nuclear Regulatory Commission was notified of the automatic actuation of the Reactor Protection System [JC].

Following the second reactor trip, it was determined (from the new data collected from instrumenting the EHC System following the first reactor trip) that the Overspeed Protection Circuit (OPC) had actuated during the system malfunction. Testing and data collected from the monitoring points revealed that there was not an actual overspeed condition; the inadvertent OPC actuation caused, or resulted from, the system malfunction. Extensive research and testing was conducted to determine the initiating event responsible for the two reactor trips from No. 13 Steam Generator low-low water level. Simulated input signals to the EHC System again revealed proper functioning of that system. No failed components were identified. Three additional printed circuit cards in the EHC System were replaced; these were two relay cards and one signal conditioning card. None of these cards could be definitely attributed to causing the event; however, they were the cards of major suspect.

APPARENT CAUSE OF OCCURRENCS:

The load rejection problem has definitely been isolated to the EHC System; however, the component responsible for the system malfunction has not yet been identified. The system is being continuously monitored to identify the root cause of the events.

The Condensate System flashing was attributed to operator error. The operator, attempting to secure three (3) heater drain pumps and two (2) condensate pumps, mistakenly secured three (3) condensate pumps and two (2) heater drain pumps.

ANALYSIS OF OCCURRENCES:

The malfunction of the EHC System on November 6, 1984, and November 11, 1984, resulted in load rejections from 91% and 93% power, respectively, to essentially no load. As a result of these sudden reductions of steam flow, steam generator pressure rapidly increased to approach no load conditions. This resulted in a shrink of the steam generator indicated level which caused the reactor trips on low-low steam generator level. The reactor trip, on low-low steam generator water level, is included in the protection system in order to preclude a loss of heat sink on a loss of feedwater. However, in these instances there was no loss of feedwater. The indicated level merely decreased due to the pressure increase which caused a collapsing of the voids in the steam producing region and to some degree in the downcomer region of the steam generator. This collapsing of voids resulted in a difference in head between the tube area and downcomer region where the level is measured. This differential pressure resulted in a rapid decrease in level in the downcomer due to more water flowing under the tube wrapper. This shrink phenomena is indistinguishable by the instrumentation from a decreasing water inventory.

During both occurrences, the Reactor Protection System functioned as designed. These events involved no undue risk to the health or safety of the public. Due to the automatic actuation of the Reactor Protection System, the events are reportable in accordance with the Code of Federal Regulations, 10CFR 50.73(a)(2)(iv).

CORRECTIVE ACTION:

As previously stated, extensive testing of the entire EHC System was performed. Some components were replaced, although none were positively identified as having initiated the events. A safety evaluation (S1-T-T600-MSE-284, Rev. 1) was performed by the Engineering Department. This evaluation concluded that the Unit could by safely operated with the EHC System in manual, and with the OPC bypassed. The mechanical overspeed circuit was thoroughly tested for proper operation, and Unit startup was authorized providing additional instrumenting of the system, continuous monitoring and operation in accordance with the safety evaluation requirements.

CORRECTIVE ACTION: (cont'd)

The Unit was returned to power operation at 1148 hours, November 22, 1984. No similar problems have been encountered to date. Operation with constant monitoring of the EHC system is continuing. Should the monitoring program identify the component responsible for these occurrences, an update report will be issued.

General Manager-
Salem Operations

JLR:tns

SORC Mtg 84-166

PSE&G

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

Salem Generating Station

December 6, 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 84-025-00

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

Sincerely yours,

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

JR/tcs

CC: Distribution

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