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 2	0 5 0 0 0 3 1 1	1 OF 0 7

TITLE (4)
Reactor Trips From 22% & 30% - High-High Level #23 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 4	2 3	8 4	8 4	0 1 0	0 0	0 5	2 3	8 4		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1
POWER LEVEL (10): 0 2 2

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	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS
X	S J	N Z L	B 0 4 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 space typewritten lines) (16)

On April 23, 1984, a turbine trip and reactor trip occurred during unit startup operations, due to high-high level in No. 23 Steam Generator. The event was attributed to sluggish response of the Feedwater Level Control System during low power operation, with minor binding of the Feedwater Control Valve Bypass Valve suspected of contributing to the magnitude of steam generator level swing. The valve was repaired and a false load was established with the Main Steam atmospheric vents during the subsequent startup. On April 27, a similar event occurred. Because the previous corrective actions had failed to remedy No. 23 Steam Generator level instability problem, the entire Feedwater Level Control System was extensively tested. The system was instrumented in order to continue the testing at low power with the turbine not latched. During this testing, on April 28, No. 23 Steam Generator Feedwater Flow indication failed to respond. The Feedwater Flow Channels were declared inoperable, and a unit shutdown was performed. Radiography revealed that the feedwater flow nozzle had moved from its designed location. This apparently occurred as a result of a feedwater water hammer incident which occurred on April 6, 1984. The feedwater flow nozzle was replaced, and all systems performed as designed during the subsequent startup on May 5, 1984.

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TEXT (If more space is required, use additional NRC Form 366A's) (17)

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 Protection System (JC) - Reactor Trips From 22% & 30% - High High Level #23 Steam Generator - (Reactor Trips #84-07 and #84-09)

Event Dates: 04/23/84
04/27/84

Report Date: 05/23/84

This report was initiated by Incident Reports 84-058 & 84-060

CONDITIONS PRIOR TO OCCURRENCES:

04/23/84 - Mode 1 - Rx Power 022% - Unit Load 0075 MWe

04/27/84 - Mode 1 - Rx Power 030% - Unit Load 0120 MWe

DESCRIPTION OF OCCURRENCES:

On April 23, 1984, unit startup operations were in progress. The generator was synchronized with the grid at 1554 hours. All Steam Generator Feedwater Level Control Systems (JB) were in automatic; and, No. 23 Steam Generator was experiencing fifteen to twenty percent (15% to 20%) level oscillations. At 1600 hours, a turbine trip and reactor trip occurred due to high-high level in No. 23 Steam Generator.

The Steam Generator Feedwater Level Control System is normally a three (3) element control system, during automatic operation. It receives signals from steam flow, feed flow and steam generator level error. At very low power levels the control system senses only the steam generator level error signal, because of the minimum steam flow and feed flow conditions. A steam generator level change has to occur before the level controller can respond. This results in sluggish response and overcompensation by the controller; and consequently, relatively large deviations from the level setpoint.

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DESCRIPTION OF OCCURRENCES: (continued)

Following this particular occurrence, 23BF19 and 23BF40 (No. 23 Steam Generator Feedwater Control Valve and Bypass Valve, respectively) were stroked. Investigation indicated that the packing was slightly "cocked" on 23BF40, causing the valve to "pop" open. It was felt that this could possibly be contributing to the magnitude of level swing associated with No. 23 Steam Generator.

Permission to perform a unit startup was given pending completion of repairs to 23BF40 packing. Instructions were also issued to the operators to establish a dummy load on the reactor using the MS10 valves (Main Steam Atmospheric Vents), and to place Steam Generator Feedwater Level Control in manual if the level was observed to spike higher than fifty percent (50%) by Narrow Range Level Indication. In addition, because of previous problems with level control during low power levels, an engineering investigation into possible future system changes was requested.

A reactor startup was performed on April 24, 1984; but a reactor trip occurred before the corrective actions taken (following the trip on April 23, 1984) could be verified to have remedied the level instability of No. 23 Steam Generator. The trip on April 24, 1984 was caused by a late opening turbine stop valve, with a subsequent momentary steam flow spike, resulting in a steam flow/feed flow mismatch with a concurrent twenty-five percent (25%) level in No. 21 Steam Generator. Since this trip was unrelated to the trip on April 23, 1984, the circumstances surrounding that occurrence are documented in LER 84-012-00.

On April 27, 1984, unit startup operations were again in progress, with all Steam Generator Feedwater Level Control Systems being monitored very closely. The generator was synchronized with the grid at 1920 hours. No. 21, 22 and 24 Steam Generator Feedwater Level Control Systems had been placed in automatic prior to synchronization. In accordance with the previous recommendations, reactor power was increased utilizing the MS10 valves. At 1923 hours, when reactor power level reached thirty percent (30%), No. 23 Steam Generator Feedwater Level Control System was placed in automatic. Level in No. 23 Steam Generator rapidly increased, resulting in a turbine and reactor trip due to high-high level in No. 23 Steam Generator. The trip occurred before the level control system could be returned to manual.

Following this occurrence, it was recognized that the corrective actions taken (following the trip on April 23, 1984) were not successful in solving the level instability of No. 23 Steam Generator. As a result, additional measures related to the entire Steam Generator Feedwater Level Control System were ordered. These actions are as follows:

DESCRIPTION OF OCCURRENCES: (continued)

A complete channel calibration procedure was performed for No. 23 Steam Generator Process Control System. The output of the valve demand controller (2FC500C) was found to be failed low. The controller would not integrate up, regardless of feed flow/steam flow mismatch and/or level error input signals. The controller was replaced, and the calibration procedure was satisfactorily completed.

Feedwater flow, steam flow and level recorders for all steam generators were calibrated. 23BF40 was stroked; the stroke was satisfactory, and its operation was smooth. No. 23 Steam Generator Process Control Loop was instrumented for subsequent startup; this included feedwater flow and steam flow process input, feedwater flow/steam flow signal summator output, level error output, controller 2FC500C output, and 23BF40 valve position. Sensor calibrations were completed on No. 23 feedwater flow (Channel I and Channel II) transmitters. A blowdown of all steam generator feedwater flow transmitter sensing lines was performed. All lines were clear with the exception of No. 23 Steam Generator Channel II. The high side transmitter line was plugged.

After reviewing the results of these investigations and corrective actions, it was concluded that the cause of the occurrence was the plugged high side sensing line of No. 23 Steam Generator Feedwater Flow Transmitter. The Station Operations Review Committee felt that the failed valve demand controller may have been a contributing factor, even though it was failed low (when discovered); which, would not have caused a high level situation. Unit startup was authorized, providing the level control systems were monitored closely for proper operation, with additional test instrumentation installed, prior to turbine latching and generator synchronization.

On April 28, 1984, a unit startup was performed. Reactor power was held at six percent (6%) while testing continued. While performing compar on checks of No. 23 Steam Generator Feedwater Control Valve (23BF19) demand signals and observing the operation of 23BF19, and also monitoring No. 23 Steam Generator level, it was discovered that feedwater flow indication failed to respond. No. 23 Steam Generator Feedwater Flow (Channels I and II) were declared inoperable. At 2330 hours Technical Specification Limiting Condition For Operation 3.0.3 was entered. At 0023 hours, April 29, 1984, the unit was placed in hot standby in accordance with the Technical Specification requirements. This occurrence is documented in LER 84-011-00.

APPARENT CAUSE OF OCCURRENCES:

As a result of the testing, which was performed during low power operation following the unit startup on April 28, 1984, it was suspected that No. 23 Feedwater Flow Nozzle (which provides the pressure drop for flow measurements used for indication, level control and protection signals) was not functioning properly; resulting in the inoperability of the feedwater flow channels. Radiography results, following the controlled shutdown, revealed that the nozzle was located approximately twenty-four inches (24") from its design location. The pins which hold the nozzle in place were apparently broken during a feedwater water hammer which occurred on April 6, 1984.

That occurrence was due to the failure of 23BF22 (No. 23 Steam Generator Feedwater Stop Check Valve) to "check" closed against steam generator pressure, while performing surveillance testing on 23BF19 (No. 23 Steam Generator Feedwater Regulating Valve). That occurrence is fully documented in Engineering Evaluation S-2-F300-MEE-021. This was determined to be the cause of the reactor trips which occurred on April 23, and April 27, 1984.

ANALYSIS OF OCCURRENCES:

The turbine trip, on high-high level in the steam generator, is an anticipatory trip. The primary function of this turbine trip is to prevent moisture carry-over, and subsequent damage to the turbine blades. The primary function of the reactor trip, on turbine trip, is to prevent steam generator safety valve actuation, due to the steam generator pressure increase, in the event that a turbine trip occurs during power operation. A turbine trip is sensed by two (2) out of three (3) signals from low autostop oil pressure or all turbine steam stop valves closed signals. A turbine trip causes a direct reactor trip above approximately ten percent (10%) reactor power (P-7 interlock circuitry), and results in a controlled short term release of steam to the turbine condenser. This steam release removes sensible heat from the RCS, and thereby avoids steam generator safety valve actuation. This reactor trip is anticipatory, and included as part of good engineering practice and prudent design. No credit is taken in any of the safety analyses for this trip. Reactor protection during startup operations is provided by the Source Range, Intermediate Range and low setting of the Power Range neutron flux trips. In both occurrences, the Reactor Protection System (JC) functioned as designed. These occurrences involved no undue risk to the health or safety of the public. Because of 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).

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TEXT (If more space is required, use additional NRC Form 366A's) (17)

CORRECTIVE ACTION:

Following the reactor trip on April 23, 1984, a review of the strip charts indicated zero feed flow to No. 23 Steam Generator; although, it was extremely hard to tell at that power level. This indication was questioned at that time, but was explained as inaccuracy of the strip chart recorders during low power operation. A reactor startup was authorized pending repairs to 23BF23 packing. In addition, special instructions were issued to the operators, and close monitoring of the level control systems was ordered.

Following the trip on April 27, 1984, a review of the strip charts revealed the same indication. Since reactor power level was at thirty percent (30%) prior to the trip, it was realized that the inaccuracy of the strip chart recorders during low power operation was no longer a plausible explanation. Although a plugged high side sensing line of No. 23 Feed Flow Transmitter was the suspected cause, further testing was scheduled to confirm these suspicions. No. 23 Steam Generator Process Control Loop was instrumented, and a unit startup was authorized, providing testing verified proper level control system operation prior to turbine latching and synchronization. When subsequent testing revealed the malfunctioning feedwater flow channels, the unit was placed in hot standby, in accordance with Technical Specification requirements.

No. 23 Feedwater Flow Nozzle was replaced with No. 13 Feedwater Flow Nozzle (from Unit 1, which is presently in a refueling outage). The feed flow transmitters associated with No. 23 Steam Generator were calibrated, utilizing the new data associated with the replacement nozzle. A unit startup was commenced at 0541 hours, May 5, 1984. Criticality was achieved at 1245 hours, and the generator was synchronized at 1731 hours. All Steam Generator Water Level Control Systems functioned as designed.

The Station Operations Review Committee (SORC) questioned the effectiveness of the post-trip review procedures. After subsequent investigations, it was determined that the post-trip review procedures were adequate and had identified a questionable feed flow indication (following the trip on April 23, 1984); however, the explanation offered was incorrect. A memorandum will be issued (with a copy of this LER attached) to all SORC members and alternates, and to those personnel involved in the post trip review. The memorandum will address the lessons learned from these events, with the intent of improving the overall quality of the post trip review process.

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LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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TEXT (If more space is required, use additional NRC Form 366A's) (17)

General Manager -
Salem Operations

JLR:kll

SORC Mtg. 84-058

NRC FORM 366A
(9-83)

PSE&G

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

Salem Generating Station

May 23, 1984

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

Dear Sir:

SALEM GENERATING STATION
LICENSE NO. DPR-75
DOCKET NO. 50-311
UNIT NO. 2
LICENSEE EVENT REPORT 84-010-00

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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