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 4

TITLE (4)
Reactor Trip From 10% Due to Low-Low Water Level in No. 23 Steam Generator

EVENT DATE (5) MONTH	DAY	YEAR	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	REPORT DATE (7) MONTH	DAY	YEAR	OTHER FACILITIES INVOLVED (8) FACILITY NAMES	DOCKET NUMBER(S)
0 7	0 8	8 5	8 5	0 1 2	0 0	0 8	0 7	8 5		0 5 0 0 0
										0 5 0 0 0

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

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 - Operations Licensing Engineer	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
B	S B	F C V	V 0 1 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 July 8, 1985, during unit startup operations, a reactor trip occurred from ten percent power level. The root cause was attributed to the lack of coordination between operators and supervisors in the control room. During Main Steam System warmup operations, the condenser steam dump valves opened rapidly upon increasing steam pressure, resulting in generator water level swings and a lowering Reactor Coolant System average temperature. One operator attempted to compensate for the lowering temperature by pulling control rods, while another operator tried to stabilize steam generator water levels utilizing the Auxiliary Feedwater System. This caused reactor power level to be raised to a point that exceeded the capacity of the Auxiliary Feedwater System, and resulted in a reactor trip on low-low water level in No. 23 Steam Generator. Extensive simulator training is being incorporated into operator training/requalification programs. A special training session is being conducted at the simulator for all licensed operators, including both supervisory and non-supervisory personnel; the emphasis of the training being placed on command and control functions and communications in the control room. Operation of the condenser steam dump control valves is being evaluated by PSE&G Engineering and the vendor.

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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 Trip From 10% During Unit Startup Operations - No. 23 Steam Generator Low-Low Water Level Signal

Event Date: 07/08/85

Report Date: 08/07/85

This report was initiated by Incicent Report No. **85-161**

CONDITIONS PRIOR TO OCCURRENCE:

Mode 1 - Rx Power 010 % - Unit Load 0000 MWe

DESCRIPTION OF OCCURRENCE:

On July 8, 1985, unit startup operations were in progress with criticality being achieved at 0321 hours. Main Steam System [SB] warmup operations were in progress, and steam generator water levels were being controlled manually via the Auxiliary Feedwater System [BA] while No. 21 Steam Generator Feed Pump (steam driven Main Feedwater Pump) warmup operations were in progress. While preparing to transfer from Auxiliary Feedwater [BA] to Main Feedwater [SJ], the condenser steam dump valves, which were in the pressure control mode of operation, opened rapidly upon increasing steam pressure. This rapid opening of the steam dump valves caused steam generator water level swings and a lowering Reactor Coolant System [AB] average temperature. One Nuclear Control Operator (NCO) attempted to stabilize the steam generator water levels with Auxiliary Feed. At the same time, a second NCO attempted to compensate for the lowering Reactor Coolant System temperature by pulling control rods [AA]. This action raised reactor power level to a point that exceeded the capacity of the Auxiliary Feedwater System (~8%). At this point, steam generator water levels began to decrease. At 0420 hours, water level in No. 23 Steam Generator reached the low-low level setpoint, resulting in a reactor trip from approximately ten percent (10%) power level.

The Unit was stabilized in Mode 3 (Hot Standby), and at 0435 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].

APPARENT CAUSE OF OCCURRENCE:

The cause of this event was personnel error, with the root cause being attributed to the lack of coordination between operators and supervisors in the control room. As previously stated, the rapid opening of the steam dump control valves caused both the steam generator water level swings and the lowering Reactor Coolant System temperature. The operators were attentive to their own areas of concern, and did not realize that their individual problems stemmed from a common cause. One operator, trying to compensate for the lowering temperature by pulling control rods, compounded the other operator's problem of trying to stabilize steam generator water levels utilizing the Auxiliary Feedwater System.

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 (2) out of three (3) 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; thus preventing steam generator dry-out and the Reactor Coolant System thermal and hydraulic transients that would be associated with a loss of the heat sink. As previously mentioned, the Auxiliary Feedwater Pumps were operating at the time of the event. The Reactor Protection System functioned as designed, and the heat sink was maintained. Since the Reactor Coolant System has been designed to withstand the thermal and hydraulic effects of four-hundred (400) reactor trips from full power, the reactor trip from ten percent (10%) power resulted in a thermal transient which 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:

Extensive simulator training is being incorporated into operator training/requalification programs; thus providing operators with the experience necessary to evaluate plant parameters, identify problem sources and take the appropriate action. Coordination and communications among operators during routine and emergency operations is being stressed. As an immediate action, an eight (8) hour training session is being conducted at the simulator for all licensed operators, including both supervisory and non-supervisory personnel; the emphasis of the training being placed on command and control functions and communications in the control room.

CORRECTIVE ACTION: (cont'd)

Because of previous problems associated with the condenser steam dump control valves, six (6) of the twelve (12) valves were replaced with ones of a different design during the last refueling outage. Design Memorandum S-C-G210-MOM-240, Rev. 1, "Replacement of Diaphragm Actuated Main Steam Bypass Valves", addresses the problem of controlling flow through the new steam dumps at low flows. Evaluations conducted with the supplier of the new valves has resulted in a proposal to modify the trim of the first group of three (3) valves to make them more responsive in low steam flows. In the interim, the atmospheric steam dumps will be used in lieu of the condenser steam dumps.

Additionally, as with all personnel error related incidents, a discussion of this event will be included in the appropriate operator training/requalification programs.

General Manager-
Salem Operations

JLR:tns

SORC Mtg 85-112

PSE&G

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

Salem Generating Station

August 7, 1985

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 85-012-00

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

Sincerely yours,

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

JLR:tcs

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