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 100% Due to Personnel Error While Testing

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
0 5	1 1	8 4	8 4	0 1 3	0 0	0 6	0 8	8 4		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)	1 0 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	

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 May 11, 1984, a reactor trip occurred due to steam flow/feed flow mismatch, coincident with No. 22 Steam Generator low water level. The trip occurred while troubleshooting and calibrating No. 22 Steam Generator Narrow Range Level Recorder. A test signal was injected into No. 22 Steam Generator level control circuitry, which simulated a high steam generator level. This caused No. 22 Feedwater Control Valve to close, resulting in a low steam generator level, and the reactor trip. The reactor trip was attributed to personnel error, due to the failure of the individual performing the task to lift the field leads from the signal isolator input for No. 22 Steam Generator Narrow Range Level Recorder prior to injecting the test signal. The root cause was attributed to inadequate procedural guidance. The individuals involved were counseled concerning their responsibility for the incident. The appropriate procedures will be modified to provide the necessary procedural guidance. In addition, the incident will be addressed in the responsible department's training program. The Reactor Protection System functioned as designed. This occurrence involved no undue risk to the health or safety of the public. Due to the automatic actuation of the Reactor Protection System, the event is reportable in accordance with 10CFR 50.73(a)(2)(iv).

IE22

NRC Form 366 (9-83)

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 100% Due to Personnel Error While Calibrating No. 22 Steam Generator Narrow Range Level Recorder

Event Date: 05/11/84

Report Date: 06/08/84

This report was initiated by Incident Report No. 84-073

CONDITIONS PRIOR TO OCCURRENCE:

Mode 1 - Rx Power 100 % - Unit Load 1150 MWe

DESCRIPTION OF OCCURRENCE:

On May 11, 1984, while in the process of calibration and troubleshooting No. 22 Steam Generator Narrow Range Level Recorder, spade lugs were installed on the input of the Signal Isolator (2LM529B) for the level recorder. A power supply was installed at this point to inject a test signal so that a calibration could be performed on the signal isolator/recorder loop. When the individual performing the work injected a 5VDC signal to verify full scale response, this signal was fed into No. 22 Steam Generator level control circuitry. This simulated a high steam generator water level signal, resulting in the closure of 22BF19 (No. 22 Steam Generator Feedwater Control Valve). The closure of 22BF19 caused a low feedwater flow; followed, twenty (20) seconds later by a low steam generator level. At 1006 hours, a reactor trip occurred, due to steam flow/feed flow mismatch, coincident with No. 22 Steam Generator low water level.

APPARENT CAUSE OF OCCURRENCE:

This reactor trip was attributed to personnel error, due to the failure of the individual performing the task to lift the field leads from the signal isolator input for No. 22 Steam Generator Narrow Range level recorder. This is required to be done before injecting a test signal to perform a recorder calibration. The following are the circumstances surrounding the occurrence.

APPARENT CAUSE OF OCCURRENCE: (cont'd)

On May 10, 1984, a Technical Assistant was assigned the job of calibrating and troubleshooting No. 22 Steam Generator level recorder, due to the erroneous indication. The responsible supervisor requested that the work be performed by applying test signals to the input of the signal isolator, in order that the complete loop be checked. The Supervisor accompanied the Technical Assistant to the work area and instructed him on the proper calibration set-up. At this time, the Supervisor explained that the signal isolator input leads had to be lifted and taped, in order that a signal would not be fed back into the control loop. After the necessary set-up was performed, the calibration and repair of the recorder was completed by the Technical Assistant. The system was then restored to normal operation.

The following day (May 11, 1984) it was observed that once again the level recorder was indicating erroneously. The same Technical Assistant was assigned the job to further investigate the problem. The responsible supervisor requested the Technical Assistant to obtain assistance from a qualified Technician to perform the necessary set-up. The Technical Assistant informed the Technician that he needed spade lugs installed on the input terminals to the signal isolator, but failed to request that the field leads be lifted. The Technician installed the lugs and left the work area. The Technical Assistant hooked up the DC power supply and injected the test signal which resulted in the reactor trip.

Subsequent interviews with the personnel involved revealed the following. The Supervisor felt that he had given sufficient instruction to the Technical Assistant on the previous day (May 10, 1984) that would enable him to inform the Tehnician of the necessary steps to be taken for the calibration set-up. This is why he did not accompany the individuals to the work area on May 11, 1984. The Technician performed only the tasks that were specifically requested. The Technical Assistant was confident with the knowledge of what had to be done, but failed to remember that the input leads were required to be lifted before the injection of the test signal.

Although the cause of the occurrence was the failure to lift the signal isolator input leads, the root cause was that proper procedural guidance was not given for the performance of this task. An individuals memory was called upon for the successful completion. The procedure utilized for this job was a general troubleshooting and calibration procedure, which gives guidelines on how to go about performing a task when no specific procedures apply. This type of procedure is necessary, in that the varied types of tasks and troubleshooting complications prohibit the use of a detailed procedure that must be followed in a step-by-step fashion.

ANALYSIS OF OCCURRENCE:

This reactor trip, on steam flow/feed flow mismatch with low steam generator water level, is an anticipatory trip. Its function is to prevent a loss of heat sink capability, by sensing conditions which would eventually result in a dry steam generator. By tripping the reactor prior to reaching the low-low level sepoint in the steam generator, the required starting time and capacity requirements for the Auxiliary Feed System [BA] are reduced; thereby, minimizing the thermal transient on the steam generators and the Reactor Coolant System [AB]. This reactor trip was caused by a false signal while testing; there were no plant conditions existing, or developing, which would have warranted actuation of the Reactor Protection System [JC]. The Reactor Protection System functioned as designed, and responded properly to the combination of signals that it saw. 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:

The individuals involved were counselled concerning their responsibility for the incident. All involved realized their failings, with the Supervisor expressing that he felt he made an error in judgement. The Technical Assistant will receive a letter of record concerning his failure to follow verbal instructions (given on May 10, 1984), which led to the occurrence.

To provide better procedural guidance for routine calibration and troubleshooting, not covered by step-by-step procedures, these procedures will be modified. This modification is to include troubleshooting/calibration check-off sheets, which will require the individual performing the task to answer pertinent questions concerning the task. These check-off sheets will require signatures to indicate that these questions have been addressed; i.e., will lifted leads be required, have all possible consequences of the task been addressed. The check-off sheets will also require Supervisor concurrence on certain answers, prior to authorization of work.

In addition, this incident will be addressed during Technician and Technical Assistant training programs. A shop letter, to all personnel within the responsible department, has been initiated, describing the circumstances which led to this inadvertent reactor trip.

John Fulton Jr.

General Manager-
Salem Operations

JLR:tns

SORC Mtg 84-066B

PSE&G

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

Salem Generating Station

June 8, 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 84-013-00

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:kil

CC: Distribution

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