PSEG

Public Service Electric and Gas Company P.O. Box 236 Hancocks Bridge, New Jersey 08038

Salem Generating Station

March 9, 1994

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

SUPPLEMENTAL LICENSEE EVENT REPORT 94-002-01

This supplemental Licensee Event Report is being submitted pursuant to Code of Federal Regulations 10CFR 50.73. It clarifies the previously submitted event analysis.

Sincerely yours,

J. J. Hagan
General Manager -
Salem Operations

MJPJ:pc

Distribution

2nd Distribution
Previously Processed
under incorrect Docket No.

180042

JE22

The power is in your hands.

95-2189 REV 7-92

NRC FORM 366 (5-92) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 5/31/95

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Salem Generating Station - Unit 2	05000 311	1 OF 04

TITLE (4)
Reactor Power Higher Than Indicated.

EVENT DATE (5)			LER NUMBER (6)			REPORT NUMBER (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
01	19	94	94	-- 002 --	01	03	09	94	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)							
			20.402(b)		20.405(c)		50.73(a)(2)(iv)		73.71(b)
POWER LEVEL (10)	100		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
			20.405(a)(1)(iii)	x	50.73(a)(2)(i)		50.73(a)(2)(viii)(A)		(Specify in Abstract below and in Text, NRC Form 366A)
			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 (Include Area Code)
M. J. Pastva, Jr. - LER Coordinator	(609) 339-5165

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 15 single-spaced typewritten lines) (16)

On 1/19/94, review of Unit 2 Fuel Cycle 8 calorimetric and Reactor Coolant System flow calculations, determined the Unit may have operated above the 3411 megawatts (thermal), specified in Operating License Condition 2.C.(1). This results from Reactor thermal power being higher than indicated by nuclear instrumentation. Preliminary data shows a potential indication error ranging from 2.5% to as high as 4.5%, resulting from feedwater flow being higher than indicated. To avoid exceeding 100% reactor power, administrative controls have been implemented to limit power to 95% by calorimetric. Nuclear instrumentation has been adjusted due to the identified error. Existing overtemperature delta temperature (OTDT) and overpower delta temperature (OPDT) setpoints provide adequate margin, as long as rod control is in manual when all rods are not fully withdrawn. The Unit will be maintained in manual rod control when all rods are not fully withdrawn until new setpoints for OTDT and OPDT are established, to reflect revised full power operating conditions. The cause of the feedwater flow indication error is under investigation. It is anticipated that on or before 3/31/94, a supplement to this report will be provided to detail results of further investigation and testing and safety significance assessment of this event.

REQUIRED NUMBER OF DIGITS/CHARACTERS FOR EACH BLOCK

BLOCK NUMBER	NUMBER OF DIGITS/CHARACTERS	TITLE
1	UP TO 46	FACILITY NAME
2	8 TOTAL 3 IN ADDITION TO 05000	DOCKET NUMBER
3	VARIES	PAGE NUMBER
4	UP TO 76	TITLE
5	6 TOTAL 2 PER BLOCK	EVENT DATE
6	7 TOTAL 2 FOR YEAR 3 FOR SEQUENTIAL NUMBER 2 FOR REVISION NUMBER	LER NUMBER
7	6 TOTAL 2 PER BLOCK	REPORT DATE
8	UP TO 18 – FACILITY NAME 8 TOTAL – DOCKET NUMBER 3 IN ADDITION TO 05000	OTHER FACILITIES INVOLVED
9	1	OPERATING MODE
10	3	POWER LEVEL
11	1 CHECK BOX THAT APPLIES	REQUIREMENTS OF 10 CFR
12	UP TO 50 FOR NAME 14 FOR TELEPHONE	LICENSEE CONTACT
13	CAUSE VARIES 2 FOR SYSTEM 4 FOR COMPONENT 4 FOR MANUFACTURER NPRDS VARIES	EACH COMPONENT FAILURE
14	1 CHECK BOX THAT APPLIES	SUPPLEMENTAL REPORT EXPECTED
15	6 TOTAL 2 PER BLOCK	EXPECTED SUBMISSION DATE

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 Power Higher Than Indicated

Event Date: 1/19/94

Supplement Report Date: 3/9/94

This report was initiated by Incident Report No. 94-027.

CONDITIONS PRIOR TO OCCURRENCE:

Mode 1 Reactor Power 100% - Unit Load 1180 MWe

DESCRIPTION OF OCCURRENCE:

On January 19, 1994, review of Unit 2 Fuel Cycle 8 calorimetric and Reactor Coolant System (RCS) flow calculations indicated that either RCS flow was low or that the Unit may have operated above the 3411 megawatts (thermal), specified in Operating License Condition 2.C.(1). Power was reduced by 3% to conservatively compensate for an estimated 2.5% error in indicated power. Preliminary data from a single feedwater flow tracer test on February 3, 1994 shows a potential indication error as high as 4.5%. To avoid exceeding 100% reactor power, administrative controls have been implemented to limit Reactor thermal power to 95% by calorimetric. In addition, nuclear instrumentation has been adjusted due to the identified error. Existing overtemperature delta temperature (OTDT) and overpower delta temperature (OPDT) setpoints provide adequate margin, as long as rod control is in manual when all rods are not fully withdrawn. The Unit will be maintained in manual rod control when all rods are not fully withdrawn until new setpoints for OTDT and OPDT are established, to reflect revised full power operating conditions.

The NRC was notified of this event per 10CFR50.72(b)(1)(ii)(B).

ANALYSIS OF OCCURRENCE:

Nuclear instrumentation trip setpoints ensure that safety limits for the reactor core and reactor coolant system are not exceeded during normal operation and design basis anticipated operational occurrences.

Review of Fuel Cycle 8 calorimetric and Reactor Coolant System flow

ANALYSIS OF OCCURRENCE: (cont'd)

calculations, shows the Unit's Operating License Condition maximum Reactor power level of 3411 megawatts (thermal) may have been exceeded. Initial assessment determined this event resulted from a potential error of 2.5% in actual Reactor thermal power higher than shown by nuclear instrumentation. Preliminary data from a single feedwater flow tracer test shows a potential indication error as high as 4.5%.

To avoid exceeding 100% reactor power, administrative controls have been implemented to limit Reactor thermal power to 95% by calorimetric. In addition, nuclear instrumentation has been adjusted for the indicated error. Evaluation of the OTDT and OPDT setpoints shows adequate margin for the existing installed values, provided there are no uncontrolled rod withdraw events. As such, the Unit will be maintained in manual rod control when all rods are not fully withdrawn. This will prevent uncontrolled rod withdraw events until new setpoints for OTDT and OPDT are established, to reflect revised full power operating conditions.

APPARENT CAUSE OF OCCURRENCE:

The cause of the feedwater flow indication error is presently under investigation.

PRIOR SIMILAR OCCURRENCES:

A review of documentation did not show any prior similar occurrence of this event.

SAFETY SIGNIFICANCE:

This is reported pursuant to the requirements of 10CFR50.73(a)(2)(i)(B) due to error introduced to the nuclear instrumentation as a result of the event.

Initial safety assessment by Westinghouse, of the potential effect of operating Salem Unit 2 at 104.5% power, shows no adverse consequence for Loss of Cooling Accidents (LOCAs). This determination was made because depending on the analysis involved, either power level is not an initial condition in the analyses or there is sufficient margin in the analyses to mitigate the effects of the event. Similarly, no adverse consequences are shown for the LOCA Containment analysis. A Salem-specific analysis, based on full power operation at 3600 MWT (WCAP 13131), has not been reviewed by the NRC and as such, is not part of the Salem licensing basis. However, the evaluation model

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SAFETY SIGNIFICANCE: (cont'd)

used for the long-term LOCA mass and energy release calculations was documented in WCAP 10325 for generic application. This model has been reviewed and approved by the NRC and has been used in the analysis of other plants.

With regard to non-LOCA events, power level is both an initial condition and a basis for the setpoints of both the Reactor Protection System and Engineered Safety Feature Actuation System. Following final assessment of the feedwater flow verification test results, the potential impact on safety significance for RPS settings and initial conditions for non-LOCA events will be assessed.

CORRECTIVE ACTION:

Administrative controls have been implemented to limit Reactor thermal power to 95% by calorimetric and nuclear instrumentation has been adjusted due to the identified error.

The Unit will be maintained in manual rod control when all rods are not fully withdrawn. This will prevent uncontrolled rod withdraw events until new setpoints for OTDT and OPDT are established, to reflect revised full power operating conditions.

It is anticipated that on or before March 31, 1994, a supplement to this report will be provided to detail the results of further investigation and testing and safety significance assessment of this event.

General Manager -
Salem Operations

MJPJ:pc

SORC Mtg. 94-021