PSEG

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

Salem Generating Station

March 30, 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 94-002-02

This supplemental Licensee Event Report is being submitted pursuant to Code of Federal Regulations 10CFR 50.73. It provides results of further event investigation, testing, and safety significance assessment.

Sincerely yours,

J. J. Hagan
General Manager -
Salem Operations

MJPJ:pc

Distribution

JE22

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

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

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.

LICENSEE EVENT REPORT (LER)

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

FACILITY NAME (1): Salem Generating Station - Unit 2

DOCKET NUMBER (2): 05000 311

PAGE (3): 1 OF 06

TITLE (4): Rx Power Higher Than Indicated & Subsequent Failure To Enter TS 3.0.3 Due To Inop. Instru.

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
01	19	94	94	-- 002 --	02	03	30	94		05000
										05000

OPERATING MODE (9): 1

POWER LEVEL (10): 100

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)
	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)	x	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: M. J. Pastva, Jr. LER Coordinator

TELEPHONE NUMBER (Include Area Code): (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)

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

EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
	06	30	94

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 (RCS) flow calculations indicated the Unit may have operated above 3411 megawatts (thermal) due to reactor thermal power greater than indicated. Power was reduced by 3% to compensate for an estimated 2.5% error in indicated power. Technical Specification 3.0.3 was not entered on 1/19/94 when Nuclear Instrumentation (NI) power range was inoperable. The NI was readjusted on 1/21/94. Data showed a potential indication error ranging from 2.5% to as high as 4.5%. Based upon the completed evaluations and results from analyses, the safety of Unit 2 was not compromised. Existing overtemperature delta temperature (OTDT) and overpower delta temperature (OPDT) setpoints provided adequate margin with manual rod control and all rods fully withdrawn. New setpoints have been established and the OTDT, OPDT, steam flow, and feedwater (FW) flow circuitry have been revised for full power operation and automatic rod control. FW instrument adjustments will be made based upon results of ongoing on line ultrasonic FW flow measurement. Results of full scale FW mock-up testing will be used to establish actual FW flow profiles. It is anticipated that by June 30, 1994, a supplement to this report will detail the results of further event investigation and testing.

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 And Subsequent Failure To Enter Technical Specification 3.0.3 Due To Inoperable Nuclear Instrumentation

Event Date: 1/19/94

Supplement Report Date: 3/30/94

This report was initiated by Incident Report Nos. 94-027 and 94-077.

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) {AB} 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.

Data from a single feedwater {SJ} flow tracer test on February 3, 1994 showed a potential indication error as high as 4.5%. To avoid exceeding 100% reactor power, administrative controls were implemented to limit Reactor thermal power to 95% by calorimetric. In addition, nuclear instrumentation (NI) {JC} was adjusted due to the identified error. Existing overtemperature delta temperature (OTDT) and overpower delta temperature (OPDT) setpoints provided adequate margin, as long as rod control was maintained in manual with all rods not fully withdrawn. The Unit was maintained in manual rod control when all rods were not fully withdrawn until new setpoints for OTDT and OPDT could be established.

New OTDT and OPDT setpoints have been established and on March 13, 1994 the OTDT and OPDT circuitry was updated to reflect revised full power operating conditions and rod control was then

DESCRIPTION OF OCCURRENCE: (cont'd)

returned to automatic. In addition, the steam flow and feedwater flow circuitry have been updated to reflect the revised full power operating conditions. On March 22, 1994, the feedwater flow nozzle flow constants in the calorimetric calculation procedure and in the on line calorimetric computer were increased, which effectively derates the Unit by 5% rated thermal power.

The NRC was notified of the potential overpower event pursuant to 10CFR50.72(b)(1)(ii)(B).

On March 3, 1994, subsequent review determined that the NI should have been readjusted on January 19 1994, following identification of the potential overpower condition. As such, the NI power range was inoperable until the NI was readjusted on January 21, 1994, and a failure to enter Technical Specification 3.0.3 occurred.

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 calculations, show 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 NI. Data from a single feedwater flow tracer test showed a potential indication error as high as 4.5%.

To avoid exceeding 100% reactor power, administrative controls were implemented to limit Reactor thermal power to 95% by calorimetric. In addition, the NI was adjusted for the indicated error. Evaluation of the OTDT and OPDT setpoints showed adequate margin for the existing installed values, provided that no uncontrolled rod withdraw events occurred. Correspondingly, the Unit was maintained in manual rod control when all rods were not fully withdrawn to prevent uncontrolled rod withdraw events.

New OTDT and OPDT setpoints have been established and the appropriate circuitry has been updated to reflect revised full power operating conditions, and rod control has been returned to automatic. In addition, steam flow and feedwater flow circuitry have been updated to reflect the revised full power operating conditions. Feedwater flow nozzle flow constants in both the calorimetric calculation procedure

ANALYSIS OF OCCURRENCE: (cont'd)

and the on line calorimetric computer were increased, which effectively derates the Unit by 5% rated thermal power.

Subsequent analysis determined the NI should have been adjusted following the conservative 3% reduction in reactor power to eliminate the possibility of operating the Unit above its licensed rated thermal power. Therefore, the NI power range was inoperable until the NI was readjusted and a failure to enter TS 3.0.3 occurred.

APPARENT CAUSE OF OCCURRENCE:

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

The failure to readjust the NI on January 19, 1994 occurred due to personnel error by Operations personnel and was a direct consequence of the immediate concern and focus to operate the Unit within its licensed rated thermal power.

PRIOR SIMILAR OCCURRENCES:

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

SAFETY SIGNIFICANCE:

This event is reportable pursuant to 10CFR50.73(a)(2)(i)(B) due the inoperability of the nuclear instrumentation as a result of the event and the subsequent failure to enter TS 3.0.3.

Initial safety assessment by Westinghouse, of the potential effect of operating Salem Unit 2 at 104.5% power, showed 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 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.

SAFETY SIGNIFICANCE: (cont'd)

Subsequent Westinghouse analysis has been performed which examined potential effects of having operated Unit 2 at power levels up to 104.5% rated power. This analysis, documented in NFSI-94-201 addressed each licensing basis LOCA and non-LOCA event and the impact of the overpower operation upon each event. For all LOCA and some non-LOCA events, engineering evaluation confirmed that no significant safety concern existed. This is because either the licensing analysis was unaffected by the overpower operation or that more than sufficient margin already existed to offset adverse consequences associated with overpower operation. For the remaining non-LOCA events, there was insufficient margin or sensitivities to assess the impact of overpower operation or to reach a conclusion without additional detailed analyses. Therefore, further analyses were performed to address these events. Based upon the completed evaluations and results from the analyses, the safety of Unit 2 was not compromised.

CORRECTIVE ACTION:

Administrative controls were implemented to limit Reactor thermal power to 95% by calorimetric and nuclear instrumentation was adjusted due to the identified error. The Unit was maintained in manual rod control when all rods were not fully withdrawn. This was done to prevent uncontrolled rod withdraw events until new setpoints for OTDT and OPDT were established, to reflect revised full power operating conditions.

The OTDT and OPDT circuitry was updated to reflect the revised full power operating conditions and rod control was returned to automatic. The steam and feedwater flow circuitry were also updated to reflect the revised full power operating conditions. The feedwater nozzle flow constants in the calorimetric calculation procedure and the on line calorimetric computer were increased by 5% to effectively derate the Unit by 5% rated thermal power, which removed the need for administrative controls on reactor power.

Adjustments will be made to the Unit feedwater system flow measurement instruments based upon the results of ongoing on line ultrasonic flow measurement of the Unit feedwater system.

In addition, the results of full scale mock-up testing of the Unit feedwater piping will be used to establish actual feedwater flow profiles.

It is anticipated that on or before June 30, 1994, a supplement to this report will be provided to detail the results of further

CORRECTIVE ACTION: (cont'd)

investigation and testing.

General Manager -
Salem Operations

MJPJ:pc
SORC Mtg. 94-026