



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

Salem Generating Station

August 12, 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 94-002-04

This supplemental Licensee Event Report is being submitted pursuant to Code of Federal Regulations 10CFR 50.73. It provides additional corrective action as well as the results of further investigation and testing.

Sincerely yours,

J. J. Hagan  
General Manager -  
Salem Operations

MJPJ:pc

Distribution

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9408230078 940812  
PDR ADOCK 05000311  
S PDR

The power is in your hands

# 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 06

TITLE (4) Reactor Power Higher Than Indicated And Subsequent Failure To Enter Technical Specification 3.0.3 Due To Inoperable Nuclear Instrumentation.

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	04	08	12	94		05000
									FACILITY NAME	DOCKET NUMBER
										05000

OPERATING MODE (9)	POWER LEVEL (10)	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR 5: (Check one or more) (11)				
I	100	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
		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: F. H. Wiltsee - LER Coordinator  
TELEPHONE NUMBER (include Area Code): (609) 339-5163

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)

<input checked="" type="checkbox"/> YES (If yes, complete EXPECTED SUBMISSION DATE)	NO	EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
			02	28	95

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 >3411 megawatts (thermal) due to reactor thermal power > 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. INITIAL Data showed a potential indication error ranging from 2.5% to 4.6%. SUBSEQUENT engineering evaluation has determined that the plant operated 1.4% > rated thermal power during Cycle 7 and 2.58% > rated thermal power during Cycle 8. Based upon completed evaluations/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. The failure to readjust the NI on 1/19/94 will be covered in Licensed Operator Regualification Training. A supplemental report detailing the cause of the feedwater flow indication error will be submitted after the completion of 2R8, presently scheduled to begin in October, 1994.

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	VARIABLES	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

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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 Power Higher Than Indicated And Subsequent Failure To Enter Technical Specification 3.0.3 Due To Inoperable Nuclear Instrumentation

Event Date: 1/19/94

Prior Submittal Date: 6/29/94

Supplement Report Date: 8/12/94

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 4.6%. 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 returned to automatic. In addition, the steam and feedwater flow circuitry have been updated to reflect the revised full power

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DESCRIPTION OF OCCURRENCE: (cont'd)

operating conditions. On March 22, 1994, the feedwater flow nozzle 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 Nuclear Regulatory Commission (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.

Subsequent engineering evaluation has determined that the plant operated 1.4% greater than rated thermal power during Cycle 7, from May 1992 through March 1993, and 2.58% greater than rated thermal power during Cycle 8, from July 1993 through January 1994.

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 4.6%.

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.

Subsequent engineering evaluation shows the plant operated 1.4% greater than rated thermal power during Cycle 7 and 2.58% greater than rated thermal power during Cycle 8.

New OTDT and OPDT setpoints have been established and the appropriate circuitry has been updated to reflect revised full power operating

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ANALYSIS OF OCCURRENCE: (cont'd)

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

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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% of rated thermal power 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 flow nozzle 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.

Ultrasonic flow measurement devices have been installed on all four FW headers and a test was conducted to determine the actual FW flow. The results of the ultrasonic feedwater flow test have been incorporated into an engineering evaluation, which determined that the plant operated 1.4% greater than rated thermal power during Cycle 7, from May 1992 through March 1993, and 2.58% greater than rated thermal power during Cycle 8, from July 1993 through January 1994.

The accuracy of the installed flow nozzles will be periodically assessed using the installed ultrasonic flow meters in conjunction with reviewing changes to plant parameters. Removal, replacement,

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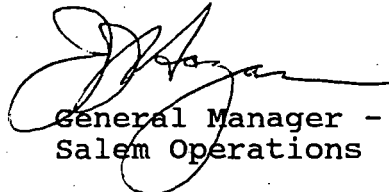
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CORRECTIVE ACTION: (cont'd)

and inspection of the nozzles will be performed during refueling outage 2R8.

The failure to readjust the NI on January 19, 1994, following the reactor power reduction, will be covered in Licensed Operator Requalification Training for 1994 - 1995.

A supplemental report detailing the cause of the feedwater flow indication error will be submitted after the completion of 2R8, presently scheduled to begin in October, 1994.

  
General Manager -  
Salem Operations

MJPJ:pc

SORC Mtg. 94-063