PSE&G

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Nuclear Business Unit

FEB 0 2 1996

LR-N96013

U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Attn: Document Control Desk

SALEM GENERATING STATION
LICENSE NO.: DPR-70 AND DPR-75
DOCKET NO. : 50-272 AND 50-311
UNIT NOS. : 1 AND 2

LICENSEE EVENT REPORT No. 94-002-06

This Licensee Event Report supplement is being submitted pursuant to the requirements of the Code of Federal Regulation 10CFR50.73. It provides additional information with respect to the root cause and corrective actions.

Sincerely,

Clay C. Warren

Clay C. Warren
General Manager -
Salem Operations

SORC Mtg. No.: 96-014

C Distribution
LER file 3.7

JE22

PSE&G makes no additional commitments in this supplemental LER.

NRC FORM 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 04/30/98
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS MANDATORY INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F33), 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.

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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 DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
									Salem station Unit 1	05000272
01	19	94	94	002	06	02	02	96	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)							
			20.2201(b)		20.2203(a)(2)(v)	X	50.73(a)(2)(i)		50.73(a)(2)(viii)
POWER LEVEL (10)	100		20.2203(a)(1)		20.2203(a)(3)(i)		50.73(a)(2)(ii)		50.73(a)(2)(x)
			20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
			20.2203(a)(2)(ii)		20.2203(a)(4)		50.73(a)(2)(iv)		OTHER
			20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below or in NRC Form 366A
			20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Michael 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
X	SJ	FI	B045	N						

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

1/19/94 review of Unit 2 Fuel Cycle 8 calorimetric and Reactor Coolant System flow calculations indicated the unit may have operated > 3411 megawatts. Power was reduced by 3% to compensate for an estimated 2.5% error in indicated power. On 1/19/94, Technical Specification 3.0.3 was not entered when Nuclear Instrumentation (NI) Power range was inoperable. The NI was readjusted on 1/21/94. Engineering evaluation determined the Unit operated 1.4% > rated thermal power during Cycle 7 and 2.58% > rated thermal power during Cycle 8. The power indication error resulted from perimeter bypass flow of the FW flow nozzles. As a result of this event the Unit 1 feedwater flow nozzles were inspected. Visual inspection showed some degradation of the nozzles. An engineering evaluation of the Unit 1 nozzles indicated a potential for having operated 0.75% greater than allowed. The unit 1 and 2 feedwater nozzles have been replaced.

PLANT AND SYSTEM IDENTIFICATION:

Westinghouse - Pressurized Water Reactor

Energy Industry Identification System (EIIS) codes appear in the text as {xx}

IDENTIFICATION OF OCCURRENCE:

Event Date: 1/19/94

Prior Submittal Date: 2/16/95

Report date: 2/2/96

Original issuance of this report initiated by Incident Report Nos. 94-027 and 94-077.

CONDITIONS PRIOR TO OCCURRENCE:

Unit 1 Reactor Power 100% - Unit Load 1180MWe

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 the Operating License Condition 2.C.(1). Power was reduced by 3% to conservatively compensate for an estimated 2.5% error in indicated power.

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 motion 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.

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DESCRIPTION OF OCCURRENCE: (cont'd):

New OTDT and OPDT setpoints were then 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 FW flow circuitry was updated to reflect the revised full power operating conditions. On March 22, 1994, the feedwater flow nozzle constants in the calorimetric computer were increased, which effectively derated 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, 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.

The Unit 1 feedwater flow nozzles were removed, examined and determined to be slightly degraded. The nozzle to pipe wall gap was large enough to allow bypass flow which eroded the carbon steel pipe, holding ring, and retaining pins. Overall, the plant parameters did not show a progressive power increase consistent with significant errors in feedwater flow measurement. Of all the parameters reviewed, the only one which may have shown an indication of a progressive power increase was RCS average differential temperature. This parameter suggests a net thermal power increase of 0.75% was possible since the apparent onset of degraded feedwater flow measurements in Unit 1 Cycle 10.

ANALYSIS OF OCCURRENCE:

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. Assessment has 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%; however, this potential error was later discounted through ultrasonic flow testing and engineering evaluation.

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ANALYSIS OF OCCURRENCE (cont'd):

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.

Following event discovery, new OTDT and OPDT setpoints were established and appropriate circuitry was updated to reflect revised full power operating conditions, and rod control was subsequently returned to automatic. In addition, steam and feedwater flow circuitry was 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 derated the Unit by 5% rated thermal power.

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.

Removal and examination of the FW flow nozzles during the Unit 2R8 refueling/maintenance outage revealed that the FW flow element inaccuracy resulted from perimeter bypass flow of the FW flow nozzles.

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 FW flow indication error was caused by perimeter bypass flow of the FW flow nozzles which resulted in erroneous differential pressure as sensed by the nozzles for a given flow rate. The bypass flow occurred due to enlargement of the annulus between each flow nozzle (stainless steel) and its respective FW pipe (carbon steel), resulting from chloride induced corrosion and subsequent erosion of the FW pipe.

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APPARENT CAUSE OF OCCURRENCE (cont'd):

The chlorides were introduced into the FW system as a result of the main turbine overspeed event on November 9, 1991 (reference LER 311/91-017-00). During the 1991 event, failed turbine blades caused multiple condenser tube ruptures, allowing aerated Delaware River water containing chlorides in excess of 3800 ppm to enter the condensate and FW systems. It is believed that following the 1991 event, an indeterminate concentration of the chlorides remained in the normally small annular crevice between the flow nozzles and the FW pipes, despite repeated fill and drain operations conducted to rinse the system during the event recovery.

Following the event and until resumption of operational FW flow in May 1992, the residual chlorides would be expected to have caused corrosion in the piping to nozzle gaps. This increased the clearances of the normally small gaps and provided a pathway for subsequent water flow, resulting in erosion of the pipe in service. This further increased the clearance of the gap to an unacceptable amount thereby providing the flow path for the incurred perimeter bypass flow.

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.

The Unit 1 feedwater flow nozzles were removed and examined. An evaluation of the impact of the Unit 1 nozzle degradation was completed and concluded that the degradation of the nozzles occurred as a result of erosion of the pipe wall at the periphery of the nozzle insert. The nozzle to pipe wall gap was large enough to allow bypass flow which eroded the carbon steel pipe, holding ring, and retaining pins.

PRIOR SIMILAR OCCURRENCE:

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

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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.

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.

The Unit 1 feedwater flow nozzle degradation is bounded by the Unit 2 evaluation described above. The health and safety of the public was not affected.

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CORRECTIVE ACTIONS:

Corrective actions taken following event discovery and prior to the Unit refueling/maintenance outage 2R8:

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 subsequently updated to reflect the revised 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%. This was done to effectively derate the Unit by 5% rated thermal power, which removed the need for administrative controls on reactor power.

Ultrasonic flow measurement devices were 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 were 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 was periodically assessed using the installed ultrasonic flow meters in conjunction with reviewing changes to plant parameters.

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

During the Unit 2R8 refueling/maintenance outage, a plant design change package (DCP) installed a precision flow measurement system in place of the failed FW flow nozzles. This system includes 1) replacement calibrated FW flow venturis without annular gaps to preclude future perimeter bypass flow and 2) precision chordal time-of-flight type ultrasonic flow meters.

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CORRECTIVE ACTIONS: (cont'd)

Startup of Unit 2 from 2R8 will be performed using the normal post refueling startup and power ascension procedures. The previous operating cycle's derated setpoints will remain in effect until new setpoints are established, based on the recently installed flow measurement devices.

PSE&G will continue to assess FW and condensate system components, potentially exposed to high chloride levels, and will take appropriate action.

A DCP, similar to the one described above for Unit 2, is being implemented in Unit 1 and will be fully installed during this outage.