PSEG

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Nuclear Business Unit

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U. S. Nuclear Regulatory Commission
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LER 272/96-013-01
SALEM GENERATING STATION - UNIT 1
FACILITY OPERATING LICENSE NO. DPR-70
DOCKET NO. 50-272

Gentlemen:

This Licensee Event Report Supplement entitled "Scaling Error of Overtemperature Delta Temperature Results in Inoperable Protection Channels" is being submitted in order to complete provide complete information for an event which was originally submitted pursuant to the requirements of the Code of Federal Regulations 10CFR50.73 (a)(2)(i)&(vii).

Sincerely,

A. C. Bakken III
General Manager
Salem Operations

Attachment

JCN/

CDistribution
LER File 3.7

NRC FORM 366 U.S. NUCLEAR REGULATORY COMMISSION
(6-1998)

LICENSEE EVENT REPORT (LER)

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

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
SALEM GENERATING STATION UNIT 1	05000272	1 OF 5

TITLE (4)
Scaling Error of Overtemperature Delta Temperature Results in Inoperable Protection Channels

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
07	11	96	96	-013	- 01	11	03	98	FACILITY NAME Salem Unit 2	DOCKET NUMBER 05000311

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR : (Check one or more) (11)

OPERATING MODE (9)	N								
			20.2201(b)		20.2203(a)(2)(v)	X	50.73(a)(2)(i)		50.73(a)(2)(viii)
POWER LEVEL (10)	000		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)	X	50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
John C. Nagle , Salem Licensing Engineer	(609) 339-3171

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX

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)

Recently, in the course of completing new instrument scaling calculations, Westinghouse notified PSE&G that the current OTDT module gain and bias setpoints could result in saturation as described in Information Notice 91-52. On July 11, 1996, PSE&G concluded that the current gain and bias settings had rendered the OTDT protection channels inoperable since the module saturation effects precluded OTDT setpoint reduction over some of the operating range. The cause of this event was investigated and found to have occurred at a vendor facility and too long ago to provide valuable root cause insight thus further investigation was not undertaken. However, contributing causes were identified which included poor vendor communications and contradicting vendor documentation. This event is reportable in accordance with 10 CFR 50.73 (a)(2)(i)(B) any operation or condition prohibited by the plant's Technical Specifications and per 10 CFR 50.73 (a)(2)(vii)(A) any event where a single cause or condition caused two independent channels to become inoperable in a single system designed to shut down the reactor. Corrective actions included revising the scaling calculations, adjusting the affected modules, performing a root cause investigation, and communicating the results of the root cause evaluation to the vendor.

TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

PLANT AND SYSTEM IDENTIFICATION

Westinghouse - Pressurized Water Reactor

Plant Protection System {JC/}*

*Energy Industry Identification System {EIIS} codes and component function identifier codes appear as (SS/CCC)

CONDITIONS PRIOR TO OCCURRENCE

At the time of identification, Salem Units 1 and 2 were shutdown and defueled.

DESCRIPTION OF OCCURRENCE

In August 1991, Information Notice 91-52 was issued which described events where improper scaling of Overtemperature Delta Temperature (OTDT) protection channels could result in the average temperature (Tavg) lead/lag compensation module saturating before the Tavg input reached the upper limit of its range. Module saturation prevents further reductions in the OTDT setpoint as Tavg continues to increase. This is contrary to the requirements for operability of the OTDT protection channels.

In 1991, PSE&G reviewed Information Notice 91-52 and concluded that there were no scaling problems of the type described in the information notice for Salem Units 1 and 2 since neither unit's OTDT channels had been adjusted as described in the information notice.

Westinghouse subsequently issued a technical bulletin in December 1991 (referenced in Supplement 1 to Information Notice 91-52), which outlined a methodology to determine whether or not OTDT hardware was scaled properly to prevent saturation during steady state and transient conditions.

Based upon the Westinghouse technical bulletin, PSE&G applied the bulletin's methodology to Salem's OTDT circuitry and determined that saturation would not occur during steady state or transient conditions during a review in 1992.

Recently, in the course of completing new instrument scaling calculations, Westinghouse notified PSE&G that the current OTDT module gain and bias setpoints could result in saturation as described in Information Notice 91-52. On July 11, 1996, PSEG concluded that the 1992 review conclusions were in error and thus the gain and bias settings had rendered the OTDT protection channels inoperable since the module saturation effects precluded OTDT setpoint reduction over some of the operating range.

CAUSE OF OCCURRENCE

A detailed Root Cause investigation was completed as a result of this event. Due to the nature of this condition and the length of time it has existed (original vendor information dating back to the 1970s), there is no value added in trying to determine a specific root cause for the noted deficiency. The investigation did identify several causal factors associated with this event as follows:

The current Salem Technical Specifications and Setpoint Uncertainty Design Basis do not agree with the vendors process equipment. The disagreement between the various vendor (Westinghouse) functions is identified as a contributing causal factor.

While this inconsistency should have been recognized by PSE&G as early as 1992, there are a number of specific times this issue should have been communicated from Westinghouse to its customers (those currently following NUREG-0452 Technical Specification format): at the time NUREG-1431 was issued (1992) and when Houston Lighting and Power issued OE 5799 (1993). As the formal communications via Westinghouse Technical Bulletins or Nuclear Safety Advisory Letters did not specifically identify this issue, this is also identified as a causal factor.

PRIOR SIMILAR OCCURRENCES

1996, 1997 and 1998 LERs were reviewed for similar occurrences. No similar events were identified.

SAFETY CONSEQUENCES AND IMPLICATIONS

The OTDT setpoints are designed to protect from violating the DNBR limit, preclude vessel exit boiling and avoid exceeding core exit quality limits of the applicable critical heat flux correlation, for the locus of operating conditions where Overpower delta temperature (OPDT) protection occurs and where the steam generator safety valves would lift. For Salem, the highest Tavg which requires OTDT protection between the OPDT setpoint and the steam generator safety valves, is less than 620 degrees F.

TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

SAFETY CONSEQUENCES AND IMPLICATIONS (continued)

For the current Salem licensing basis analysis setpoints, this represents the point where the high pressure safety analysis OTDT setpoint equation line intersects the steam generator safety valve line. For the Fuel Upgrade/Margin Recovery Program (FU/MRP) analysis setpoints, this intersection point is also at a Tavg of less than 620 degrees F. Under steady state conditions, the fact that the OTDT setpoint saturates at Tavg greater than 630 degrees F (i.e. the setpoint is not reduced) is not a concern because it is beyond the range of Tavg that the OTDT setpoint is required to protect.

The following discusses the impact of the scaling issue for transient conditions. The purpose of the lead/lag module is to ensure that the OTDT setpoint trips the reactor before the actual conditions that would cause DNB or vessel exit boiling are reached. For slow transients (i.e., events that result in a slow increase in the indicated Tavg) the output of the Tavg lead/lag module will not significantly lead the input indicated Tavg and a reactor trip will occur prior to the output of the lead/lag module saturating (the Salem OTDT setpoint only needs to provide protection to an indicated Tavg of 620 degrees F). In the event of a fast transient (i.e., an event that results in a rapid increase in the indicated Tavg) the output of the lead/lag module could saturate when the indicated Tavg is well below a Tavg of 620 degrees F, especially for lower power levels where a higher Tavg does not result in a reactor trip. However, it is important to note that while the OTDT setpoint would not be reduced, it is the indicated Tavg (essentially the actual Tavg) that determines the margin to DNB and vessel exit boiling limits. In addition, at the power levels where this is more of a concern, there is significant margin between the OTDT setpoints and the DNB safety limits. Therefore, the fact that the Tavg lead/lag module saturates is not a safety concern because there would be significant margin between the indicated/actual conditions and those conditions that define the actual safety limits.

The corrective actions described below, performed by Westinghouse and PSE&G Nuclear Fuels group to assess the impacts of the lead-lag setting tolerance on the accident analysis determined that:

1. For accidents in which DNBR is the acceptance criteria, there is either no change to DNBR margin or minimal impact (well less than available excess margin). This covers lead-lag setting tolerance for the OTDT trip and rod control. Over Pressure delta Temperature OPDT is not considered here since its not credited in the analysis.

FACILITY NAME (1)	DOCKET (2) NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Salem Unit 1	05000272	96	-- 013 --	01	5 OF 5

TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

SAFETY CONSEQUENCES AND IMPLICATIONS (continued)

2. For steamline break accidents, the effects of the lead-lag setting tolerance on the high steam flow/low steam pressure safety injection and steamline isolation is inconsequential with respect to core response or containment design limits.

In summary, the current time constant setting/calibration process will not adversely impact the results of the design basis accident analyses.

There are no safety consequences for this occurrence and the safety and health of the public were not affected.

CORRECTIVE ACTIONS

1)The scaling calculations were revised and the modules were adjusted prior to each Unit entering mode 2.

2)To verify the lead-lag setpoint tolerance is covered within the accident analysis margins: (a) Westinghouse performed an analysis and provided a letter which concluded that those accidents which rely on the OTDT trip are modeled sufficiently conservative, such that assuming nominal dynamic terms introduces no unacceptable consequences. The conclusion of this assessment is also applicable to other dynamically compensated protective functions. This is a compensatory action which allowed OT and OPDT loop calibration to continue with the current tolerance range. (b) PSE&G Nuclear Fuels group performed an in-house assessment (documented in controlled calculation files) of all lead-lag functions assumed in the accident analyses. This action, which validated and expanded upon the Westinghouse position, concluded that the tolerance on the nominal lead-lag values has inconsequential impact to safety margins.

3) Those items identified as causal factors are based on Westinghouse's programs and processes. As a customer, PSE&G has limited control over these other than feedback through QA audits, technical reviews, responses to NRC and other industry and user groups. For this reason there are no specific corrective actions assigned to correct these causes. However, with respect to the issue at hand, PSE&G Nuclear Fuels group has transmitted a copy of the completed root cause evaluation to Westinghouse. This will serve as feedback tool for their customer notification process, and can be considered a corrective action to prevent recurrence.

4) The accident analysis input assumption database which was developed as part of the SALEM UFSAR review project has been reviewed and revised for the items under reactor protection system such that lead-lag terms reflect a +/-10% setting tolerance.