PSEG Nuclear LLC
P.O. Box 236, Hancocks Bridge, New Jersey 08038-0236

PSEG
Nuclear LLC

OCT 2 0 2000
LR-N000417

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

LER 272/00-004-00
SALEM GENERATING STATION - UNIT 1
FACILITY OPERATING LICENSE NO. DPR-70
DOCKET NO. 50-272

Gentlemen:

This Licensee Event Report entitled "Failure to Comply with Technical Specification 3.7.10 Action a.1" is being submitted pursuant to the requirements of 10CFR50.73(a)(2)(i)(B).

Sincerely,

D. F. Garchow
Vice President -
Technical Support

Attachment

BJT

C Distribution
LER File 3.7

IE22

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)

APPROVED BY OMB NO. 3150-0104 EXPIRES 06/30/2001

Estimated burden per response to comply with this mandatory information collection request: 50 hrs. Reported lessons learned are incorporated into the licensing process and fed back to industry. Forward comments regarding burden estimate to the 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. If an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
SALEM UNIT 1	05000272	1 OF 4

TITLE (4)
Failure to Comply with Technical Specification 3.7.10 Action a.1

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
09	25	00	00	-004	- 00	10	20	00	Salem 2	05000311
									FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	1
POWER LEVEL (10)	~35

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)
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)
Brian Thomas, Licensing Engineer	(856) 339-2022

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)

On September 25, 2000, during the revision of an engineering evaluation for the chilled water system, engineering identified an invalid assumption associated with the determination of heat loads to remove from the chilled water system {KM/CHU}. Due to the invalid assumptions in the engineering evaluation, the table provided in the engineering evaluation for the amount of heat loads to remove from the system based on the chiller cooling water inlet temperature provided non-conservative heat load values. Due to the non-conservative heat loads, the operations procedures did not specify the removal of enough of the appropriate non-essential heat loads from the chilled water system {KM/CHU} to meet TSAS 3.7.10.a.1. Although TSAS 3.7.10.a.1 was not complied with, the chilled water system would have performed as required during a design basis accident.

The apparent cause of this event is attributed to personnel errors. The engineering personnel involved in the development of the engineering evaluation did not validate the assumption used for chiller operation and the accuracy of the formula used to calculate the heat loads. Also, the verifier of the engineering evaluation did not identify the above errors.

The engineering evaluation and operations procedures have been revised to correct the non-conservative heat load values. Appropriate personnel were held accountable for their actions and lessons learned from this event will be communicated to the appropriate Engineering personnel. This report is being submitted in accordance with 10 CFR 50.73(a)(2)(I)(B)

FACILITY NAME (1)	DOCKET (2) NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
SALEM UNIT 1	05000272	00	0 0 4	00	2 OF 4

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

PLANT AND SYSTEM IDENTIFICATION

Westinghouse – Pressurized Water Reactor

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

CONDITIONS PRIOR TO OCCURRENCE

At time of discovery, Salem Unit 1 was in Mode 1 with reactor power at approximately 35%, Salem Unit 2 was in Mode 1 with reactor power at approximately 90% and decreasing towards the beginning of its 11th refueling outage. No structures, systems, or components were inoperable at the time of the occurrence that contributed to the event.

DESCRIPTION OF OCCURRENCE

On September 25, 2000, during the revision of an engineering evaluation for the chilled water system {KM/CHU}, Engineering identified an invalid assumption associated with the determination of heat loads to remove from the chilled water system {KM/CHU}. The results of the engineering evaluation were incorporated into operations procedures S1(2).OP-SO.CH-0001 for operation of the chilled water system in November 1998. Technical Specification 3.7.10 action statement a.1 states that with one chiller inoperable:

> "Remove the appropriate non-essential heat loads from the chilled water system within 4 hours and;..."

Due to the invalid assumptions in the engineering evaluation, the table provided in the engineering evaluation for the amount of heat loads to remove from the system based on the chiller cooling water inlet temperature provided non-conservative heat load values. Due to the non-conservative heat loads, the operations procedures did not specify the removal of enough of the appropriate non-essential heat loads from the chilled water system {KM/CHU} to meet TSAS 3.7.10.a.1. The TSAS for one inoperable chiller has been entered in the past using procedures S1(2).OP-SO.CH-0001 which contained the non-conservative heat loads. This is reportable in accordance with 10CFR50.73(a)(2)(i)(B), as a technical specification prohibited condition. Although TSAS 3.7.10.a.1 was not complied with, the chilled water system would have performed as required during a design basis accident.

FACILITY NAME (1)	DOCKET (2) NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
SALEM UNIT 1	05000272	00	0 0 4	00	3 OF 4

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

ANALYSIS OF OCCURRENCE

On September 25, 2000, engineering personnel identified that an assumption used in evaluation S-C-CH-MEE-1243, "Engineering Evaluation of Auxiliary Building Chilled Water Subsystem Loading – Salem Units 1 and 2," Revision 1 was incorrect. This engineering evaluation was used to revise operations procedures S1(2).OP-SO.CH-0001, "Chilled Water System Operation," to provide tables to be used by the operators in determining what heat loads to remove from the chilled water system {KM/CHU} to comply with TSAS 3.7.10.a.1 when a chiller was determined to be inoperable. Procedures S1(2).OP-SO.CH-0001 were revised on November 25, 1998. Two tables were added to the procedure based on the engineering evaluation. The first table provided a range of chiller cooling water inlet temperatures and the associated amount of heat loads to remove from the system if one or two chillers were out of service. The second table provided a listing of the equipment considered non-essential loads and the value of heat load that would be removed from the chilled water system when that piece of equipment was isolated.

The first table provided a range of chiller cooling water inlet temperatures from <60 °F to 85 °F in increments of 5 degrees based on the assumption that the temperature of the cooling water was the same temperature of the water entering the chiller. Upon further review of the chiller system operation, engineering identified that the assumption that the water entering the chiller was the same temperature as the cooling water supply was incorrect. The chillers are provided with a recirculation pump that is currently set to maintain the chiller cooling water inlet temperature at a minimum of 68 °F. Therefore, the heat load removal specified for temperatures less the 68 °F was in error.

A second error was also identified in the evaluation concerning the amount of heat loads to be removed based on chiller cooling water inlet temperature. The calculation performed to determine the heat loads to remove contained an error. For chiller cooling water inlet temperatures of ≤ 80 °F, the amount of heat loads calculated to be removed were non-conservative.

CAUSE OF OCCURRENCE

The apparent cause of this event is attributed to personnel errors. The engineering personnel involved in the development of the engineering evaluation did not validate the assumption used for chiller operation and the accuracy of the formula used to calculate the heat loads. Also, the verifier of the engineering evaluation did not identify the above errors.

PRIOR SIMILAR OCCURRENCES

A review of 1998 and 1999 LERs for both Salem and Hope Creek identified no similar instances of engineering personnel errors resulting in the failure to comply with the Technical Specifications.

FACILITY NAME (1)	DOCKET (2) NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
SALEM UNIT 1	05000272	00	0 0 4	00	4 OF 4

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

SAFETY CONSEQUENCES AND IMPLICATIONS

There were no safety consequences or implications associated with this event. Although the appropriate non-essential heat loads were not removed as required by TSAS 3.7.10.a.1, during an accident or loss of offsite power (LOOP), the non-essential heat loads are isolated automatically from the Unit's chilled water system by the initiation of the safety injection or LOOP signal. Therefore, the chilled water system would have performed as required during a design basis accident.

A review of this event determined that a Safety System Functional Failure (SSFF) as defined in NEI 99-02 did not occur.

CORRECTIVE ACTIONS

1. Engineering evaluation S-C-CH-MEE-1243 was revised on September 26, 2000.
2. Procedures S1(2).OP-SO.CH-0001, "Chilled Water System Operation," were revised on September 29, 2000, to incorporate the amount of heat loads to remove from the chilled water system as specified in the revised engineering evaluation.
3. Appropriate personnel were held accountable for their actions.
4. Lessons learned from this event will be communicated to the appropriate Engineering personnel and will be implemented under PSEG Nuclear's corrective action program.

COMMITMENTS

The corrective actions cited in this LER are voluntary enhancements and do not constitute commitments.