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Public Service Electric and Gas Company P.O. Box 236 Hancocks Bridge, New Jersey 08038-0236

Nuclear Business Unit

JUL 07 1997

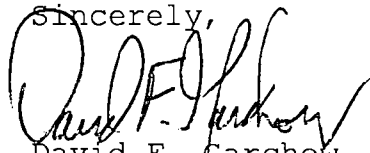
LR-N970421

U. S. Nuclear Regulatory Commission
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Washington, DC 20555

Gentlemen:

LER 272/96-008-01
SALEM GENERATING STATION - UNIT 1
FACILITY OPERATING LICENSE NO. DPR-70
DOCKET NO. 50-272

This Supplemental Licensee Event Report entitled "Failure to Meet Technical Specifications Requirement While in Mode 5 for Natural Circulation" is being submitted pursuant to the requirements of the Code of Federal Regulations 10CFR50.73(a)(2)(i)(B).

Sincerely,

David F. Garchow
General Manager -
Salem Operations

Attachment

BJT

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LER File 3.7

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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 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.

FACILITY NAME (1)
SALEM GENERATING STATION, UNIT 1

DOCKET NUMBER (2)
05000272

PAGE (3)
1 OF 4

TITLE (4)
Failure to Meet Technical Specifications Requirement While in Mode 5 for Natural Circulation

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
06	13	96	96	-- 008	-- 01	07		97	SALEM UNIT 2	05000311
									FACILITY NAME	DOCKET NUMBER
										05000

OPERATING MODE (9)	N	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)					
POWER LEVEL (10)	0	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: Brian J. Thomas, Licensing Engineer
TELEPHONE NUMBER (Include Area Code): 609-339-2022

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)

YES (If yes, complete EXPECTED SUBMISSION DATE): X NO

EXPECTED SUBMISSION DATE (15)

MONTH	DAY	YEAR

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On June 13, 1996, a review identified that Salem Units 1 and 2 had operated in the past in Mode 5 in a condition not in compliance with Technical Specification 3.4.1.4. Specifically, the units were operated with a Residual Heat Removal loop inoperable when the Reactor Coolant System loops may not have been adequately filled to support natural circulation. This issue was identified during reviews as a result of NRC Information Notice 95-35. On June 6, 1997, further review determined that during past performances of Manual Safety Injection testing, both RHR loops on Unit 2 were rendered inoperable while the unit was in Mode 5, which was not in compliance with TS 3.4.1.4. None of the past conditions identified occurred after issuance of Information Notice 95-35.

The cause of this occurrence is attributed to insufficient technical input for the development of controls for relying on steam generators as a backup method for decay heat removal. Additional controls were subsequently put in place to ensure the availability of natural circulation in Mode 5 with one RHR loop inoperable.

This event is reportable in accordance with 10 CFR 73(a)(2)(i)(B), any condition prohibited by the plant's Technical Specifications.

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TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
SALEM GENERATING STATION, UNIT 1	05000272	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	2 OF 4
		96	-- 008	-- 01	

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

PLANT AND SYSTEM IDENTIFICATION

Westinghouse - Pressurized Water Reactor

Residual Heat Removal System (RHR) {BP/-}*

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

CONDITIONS PRIOR TO OCCURRENCE

The plant was in Mode 5 prior to each event.

DESCRIPTION OF OCCURRENCE

On June 13, 1996, a review identified that Salem Units 1 and 2 had operated in the past in Mode 5 in a condition not in compliance with Technical Specification 3.4.1.4. Specifically, the units were operated with a Residual Heat Removal (RHR) loop inoperable when the Reactor Coolant System (RCS) {AB/-} loops may not have been adequately filled to support natural circulation. Salem Technical Specifications 3.4.1.4, Reactor Coolant System (RCS), states that when in Mode 5, "Two residual heat removal loops shall be OPERABLE and at least one RHR loop shall be in operation." A note to this section states that, "One RHR loop may be inoperable for up to two hours for surveillance testing, provided the other RHR loop is OPERABLE and in operation. Additionally, four filled reactor coolant loops, with at least two steam generators with their secondary side water levels greater than or equal to 5 percent (narrow range), may be substituted for one residual heat removal loop."

NRC Information Notice 95-35 addressed the potential to degrade the ability of steam generators to remove decay heat by natural circulation. The Information Notice indicated that if the RCS was vented, decay heat removal by natural circulation may not occur. This condition had previously been identified at Vogtle Unit 1 and Turkey Point Units 3 and 4. As a result of the NRC Information Notice, a condition report was initiated in the Corrective Action Program to review this issue for Salem Units 1 and 2. A subsequent engineering evaluation for this issue stated that a minimum RCS pressure of 100 psig must be maintained to take credit for natural circulation. The review completed on June 13, 1996 found occasions where in the past, one RHR loop was inoperable and the RCS pressure was below 100 psig. Additional controls were subsequently put in place to ensure the availability of natural circulation in Mode 5 with one RHR loop inoperable.

On June 6, 1997, a different review was performed which evaluated past performances of the Unit 2 Manual Safety Injection surveillance test. The review determined that both loops of RHR were rendered inoperable during the test of the semi-automatic RHR suction swapover circuitry. The test format secured both RHR pumps and closed the isolation valves in the common suction

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

DESCRIPTION OF OCCURRENCE (cont'd)

line. The review of historical records indicated that although testing had been performed on Salem Unit 2 while the Steam Generators met the requirements for RHR loop substitution (i.e., four filled reactor coolant loops, with at least two steam generators with their secondary side water levels greater than or equal to 5 percent narrow range), RCS pressure was less than 100 psig during the testing which did not comply with the requirement of TS 3.4.1.4. The most recent occurrence identified was December 17, 1994. Salem Unit 1 does not have the RHR suction swapover circuitry. The comparable Unit 1 test does not concurrently shutdown both RHR pumps nor close the RHR suction valves, and thus does not result in failure to comply with TS 3.4.1.4.

CAUSE OF OCCURRENCE

The cause of this occurrence is attributed to insufficient technical input for the development of controls for relying on steam generators as a backup method for decay heat removal.

PRIOR SIMILAR OCCURRENCES

A review of LERs for Salem Units 1 and 2 over the last two years identified one LER that was a result of an industry problem identified in a NRC Information Notice or Generic Letter. LER 272/96-002-01 addressed a concern for motor operated gate valves that were susceptible to pressure locking and thermal binding. This issue was identified in NRC Generic Letter 95-07. Corrective actions were specific to the susceptible valves.

SAFETY SIGNIFICANCE

During the period that the steam generators were credited as a decay heat removal source, one train of RHR was available for removal of decay heat. On the loss of the operating RHR loop, the operators would enter the abnormal procedure for loss of RHR and take the appropriate actions. A review of the evaluated occasions determined that there were no challenges to the operating RHR loop and, therefore, no actual safety consequences. The health and safety of the public was not affected.

In the instances where both loops of RHR were rendered inoperable during Manual Safety Injection testing, the actions which rendered RHR inoperable were controlled by a procedure, with test personnel fully aware of the requirements of TS 3.4.1.4 in regard to monitoring RCS temperature and boron concentration. TS 3.4.1.4 permits both RHR pumps to be off for up to 2 hours so long as RCS temperature and boron concentration are monitored. Although the monitoring of temperature and boron concentration met the requirement of an RHR loop in operation, the requirement for an OPERABLE RHR loop was not met. The amount of time the RHR loops were inoperable was minimal (less than two hours) as determined by narrative log entries. There were no direct challenges to reactor safety presented by the occurrences. The health and safety of the public were not affected.

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

CORRECTIVE ACTIONS

1. Plant operating logs for Mode 5 have been revised to include specific criteria to ensure that the Reactor Coolant System loops are full and natural circulation conditions can be established if steam generators are being relied on for a backup to an inoperable RHR system.
2. RHR surveillance procedures for Unit 1 and 2 were reviewed for additional concerns with inadvertent failures to comply with Technical Specification requirements resulting from test alignments. No further deficiencies were identified by this review.
3. The Unit 2 surveillance test procedure will be revised to address the requirements of Technical Specification 3.4.1.4. This procedure will be revised prior to the next performance of the surveillance test.