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

LER 272/02-009-00
SALEM GENERATING STATION - UNIT 1
FACILITY OPERATING LICENSE NO. DPR-70
DOCKET NO. 50-272

Gentlemen:

This Licensee Event Report entitled "Failure To Perform Required Action of Technical Specification 3.1.3.2.1" is being submitted pursuant to the requirements of 10CFR50.73(a)(2)(i)(B).

Sincerely,

L. Waldinger
Director – Site Operations

Attachment

BJT

C Distribution
LER File 3.7

IE22

NRC FORM 366 (7-2001)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

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

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
SALEM GENERATING STATION UNIT 1	05000272	1 OF 4

4. TITLE

FAILURE TO PERFORM REQUIRED ACTION OF TECHNICAL SPECIFICATION 3.1.3.2.1.

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MO	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
									Salem Unit 1	
12	12	02	02	– 009 –	00	02	10	03	FACILITY NAME	DOCKET NUMBER

9. OPERATING MODE	1
10. POWER LEVEL	100

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply)

	20 2201(b)		20 2203(a)(3)(ii)		50.73(a)(2)(ii)(B)		50 73(a)(2)(ix)(A)
	20.2201(d)		20 2203(a)(4)		50.73(a)(2)(iii)		50.73(a)(2)(x)
	20.2203(a)(1)		50 36(c)(1)(i)(A)		50.73(a)(2)(iv)(A)		73 71(a)(4)
	20.2203(a)(2)(i)		50 36(c)(1)(ii)(A)		50.73(a)(2)(v)(A)		73.71(a)(5)
	20.2203(a)(2)(ii)		50 36(c)(2)		50.73(a)(2)(v)(B)		OTHER Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iii)		50 46(a)(3)(ii)		50.73(a)(2)(v)(C)		
	20.2203(a)(2)(iv)		50 73(a)(2)(i)(A)		50.73(a)(2)(v)(D)		
	20.2203(a)(2)(v)	X	50.73(a)(2)(i)(B)		50.73(a)(2)(vii)		
	20.2203(a)(2)(vi)		50 73(a)(2)(i)(C)		50.73(a)(2)(viii)(A)		
	20.2203(a)(3)(i)		50 73(a)(2)(ii)(A)		50.73(a)(2)(viii)(B)		

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER (Include Area Code)
Brian J. Thomas, Licensing Engineer	856-339-2022

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

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX

14. SUPPLEMENTAL REPORT EXPECTED				15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE)	X	NO				

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

On December 12, 2002, at 0821, the individual rod position indication (IRPI) for control rod 1C3 was declared inoperable on Salem Unit 1. Technical Specification action statement (TSAS) 3.1.3.2.1.a was entered which requires that either the position of the non-indicating rod is to be determined by use of the power distribution monitoring system or incore movable detectors once every 8 hours or to reduce thermal power to less than 50% of rated thermal power. The verification of the position of rod 1C3 was performed at 1430 and 2030 hours using the power distribution monitoring system. However, upon subsequent review of the 1430 and 2030 performance, the duty reactor engineer identified that the rod position verification had been performed on Unit 2 instead of Unit 1.

The causes of this event are attributed to imprecise communication between the Control Room Supervisor (licensed operator) and Reactor Engineer due to failure to conduct a pre-job brief and failure of personnel to utilize human error reduction techniques. A contributing cause to this event was inadequate procedure guidance for the Reactor Engineers. Personnel involved in this event from Operations and Reactor Engineering have been held accountable in accordance with company policies. Additionally this event will be evaluated for inclusion in Licensed Operator Training Program and Engineering Continuing Training Program, lessons learned will be communicated to the engineering organization and Reactor Engineering procedures have been revised.

This report is being made in accordance with 10CFR50.73(a)(2)(i)(B), "any operation or condition which was prohibited by the plant's Technical Specification."

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

PLANT AND SYSTEM IDENTIFICATION

Westinghouse – Pressurized Water Reactor

Incore Monitoring System (IG/-)

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

CONDITIONS PRIOR TO OCCURRENCE

Salem Unit 1 was in Mode 1 at 100% power at the time of discovery. The individual rod position indication (IRPI) for control rod 1C3 was inoperable. No additional equipment was out of service that contributed to this event.

DESCRIPTION OF OCCURRENCE

On December 12, 2002, at 0821, the individual rod position indication (IRPI) for control rod 1C3 was declared inoperable on Salem Unit 1. Technical Specification action statement (TSAS) 3.1.3.2.1 states:

"a. With a maximum of one analog rod position indicator per bank inoperable either:

1. Determine the position of the non-indicating rod(s) indirectly using the power distribution monitoring system (if power is above 25% RTP) or using the movable incore detectors (if power is less than 25% RTP or the power distribution monitoring system is inoperable) at least once per 8 hours and within one hour after any motion of the non-indicating rod which exceeds 24 steps in one direction since the last determination of the rod's position, or

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 8 hours."

At 1330 hours, the Unit 1 Control Room Supervisor (CRS) informed the duty reactor engineer that the IRPI for rod 1C3 had been declared inoperable and as required by TSAS 3.1.3.2.1.a.1, the rod position needed to be verified every 8 hours using the power distribution monitoring system (IG/-). The first verification of the position of rod 1C3 using the power distribution monitoring system was completed at 1430 hours. The duty reactor engineer determined the next surveillance would be due at 2030 hours (6 hours from last performance) to ensure that the 8 hour action requirement is met. At 2030 hours, the duty reactor engineer performed the next verification and provided the information to the Unit 1 CRS for review. After providing the rod position verification information to the Unit 1 CRS, TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

DESCRIPTION OF OCCURRENCE (cont'd)

the duty reactor engineer subsequently identified that the rod position verifications performed at 1430 and 2030 were performed on the wrong unit. The position of rod 1C3 had been verified on Salem Unit 2 instead of Salem Unit 1. The duty reactor engineer immediately called the Unit 1 CRS and informed the CRS of the rod position verification errors. Technical Specification 3.0.3 was entered at 2100 hours based on the missed action requirements. The Unit 1 CRS and the Operations Superintendent (OS) directed the duty reactor engineer to perform the rod position verification for rod 1C3 on Salem Unit 1. The rod position verification for Unit 1 rod 1C3 was initiated at 2140 hours and completed at 2330 hours utilizing the power distribution monitoring system. At 2330 hours TS 3.0.3 was exited.

This LER is being submitted pursuant to 10CFR50.72(a)(2)(i)(B) for any operation or condition which was prohibited by the plant's Technical Specifications.

CAUSE OF OCCURRENCE

The causes of this event are attributed to imprecise communication between the Control Room Supervisor (licensed operator) and Reactor Engineer due to failure to conduct a pre-job brief and failure of personnel to utilize human error reduction techniques. Communication between the CRS (licensed operator) and the duty reactor engineer was incomplete in that it was not validated that the duty reactor engineer understood which unit that the rod position verification was to be performed on. Proper use of human error reduction techniques (i.e., STAR (Stop, Think, Act Review), QV&V (Qualify, Validate and Verify) and two clarifying questions) were not exhibited by the CRS and the duty reactor engineer. The first rod position verification performed by the duty reactor engineer was verified by a second engineer. However instead of validating the correct Unit for performance of the rod position verification, the second reactor engineer relied on the first reactor engineer stating that the verification was for rod 1C3 in Unit 2 (instead of Unit 1). During the acceptance review of the rod position verification actions, the Unit 1 CRS did not validate that the verification was performed on the correct unit. A contributing cause to this event was inadequate procedure guidance for the Reactor Engineers.

PRIOR SIMILAR OCCURRENCES

A review of LERs for Salem and Hope Creek for the previous two years did not identify any similar occurrences associated with a missed Technical Specification action due to performance of the action on the wrong unit.

SAFETY CONSEQUENCES

The purpose of Technical Specification 3.1.3.2.1 is to ensure that the control rod position indication system and position demand system are operating properly. Control rods are verified to be within a certain number of steps of the demanded position by use of the individual rod position indication (IRPI) and group demand counters. Verification that rods are within the required number of steps of the demanded position provides assurance that the core reactivity parameters are within the bounds used in the accident analysis. Although the IRPI was inoperable for rod 1C3, the rod position was subsequently validated to be within the required number of steps of the group demand counter providing confirmation that the core reactivity parameters were maintained within the bounds of the accident analysis. Therefore there were no safety consequences associated with this event.

A review of this event determined that a Safety System Functional Failure (SSFF) as defined in Nuclear Energy Institute (NEI) 99-02 has not occurred.

CORRECTIVE ACTIONS:

1. Personnel involved in this event from Operations and Reactor Engineering have been held accountable in accordance with company policies.
2. Lessons learned from this event will be provided to the engineering organization. Additionally, this event will be reviewed, including the failure to utilize pre-job briefs, by the Engineering Training Review Group to evaluate changes to the Engineering Continuing Training Program.
3. This event will be reviewed, including the failure to utilize pre-job briefs, by the Operations Training Review Group to evaluate changes to the Licensed Operator Training Program.
4. Reactor Engineering procedures used to perform rod position verification, SC.RE-RA.RCS-0017(Q) and SC.RE-SO.NIS-0001(Q) have been revised to include requirements to conduct a pre-job brief and use pertinent human error reduction techniques (e.g., QV&V, STAR, 3-point communication).

The above actions are being tracked in accordance with PSEG Nuclear's corrective action program.

COMMITMENTS

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