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

LER 311/03-003-00
SALEM GENERATING STATION - UNIT 2
FACILITY OPERATING LICENSE NO. DPR-75
DOCKET NO. 50-311

Gentlemen:

This Licensee Event Report entitled "Manual Reactor Trip Due To Dropped Control Rod," is being submitted pursuant to the requirements of 10CFR50.73 (a)(2)(iv)(A).

Sincerely,

C. Fricker
Plant Manager - Salem

Attachment

BJT

C Distribution
LER File 3.7

IE22

NRC FORM 366 (7-2001) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104 EXPIRES 7-31-2004

Estimated burden per response to comply with this mandatory information collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records Management Branch (T-6 E6), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by Internet e-mail to bjs1@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose 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 collection.

LICENSEE EVENT REPORT (LER)

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

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
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4. TITLE

MANUAL REACTOR TRIP DUE TO DROPPED CONTROL ROD

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
11	23	03	03	– 003	– 00	01	20	04	FACILITY NAME	DOCKET NUMBER

9. OPERATING MODE	2
10. POWER LEVEL	0

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)	X 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)	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
B	AA	FU	W120	Y					

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 November 23, 2003, during low power physics testing with control bank D control rods at 209 steps, control rod 1D4 dropped in the reactor core. At 0507 hours, the Operating crew entered the abnormal procedure for a dropped control rod. Based upon the dropped rod causing the reactor to go subcritical, the abnormal procedure directs all rods to be inserted. The reactor was manually tripped as directed by the Control Room Supervisor. The unit was stabilized in Mode 3 with the reactor at normal operating temperature and pressure with core cooling provided by the auxiliary feedwater system and main steam dumps to the condenser.

The cause of the dropped rod and subsequent manual reactor trip is attributed to a blown fuse (10 amp) for control rod 1D4. The fuse blew as a result of infant mortality. In addition to the infant mortality, the present 10 amp fuses in the rod control system have minimal margin to prevent failure of the fuse during maximum peak current periods based on a study performed by the vendor of the rod control system. This minimal margin may have also contributed to the fuse blowing. A design change was implemented to replace the existing 10 amp fuses with higher amperage fuses in the Unit 2 rod control system.

This report is being made in accordance with 10CFR50.73(a)(2)(iv)(A), "any event or condition that resulted in manual or automatic actuation of any of the systems listed in paragraph (a)(2)(iv)(B)."

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

PLANT AND SYSTEM IDENTIFICATION

Westinghouse – Pressurized Water Reactor

Rod Control System (AA/-)

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

CONDITIONS PRIOR TO OCCURRENCE

Salem Unit 2 was in Mode 2 at approximately 0% reactor power performing low power physics testing at the time the reactor was manually tripped. No additional equipment was out of service that contributed to this event.

DESCRIPTION OF OCCURRENCE

On November 23, 2003, with control bank D control rods at 209 steps, control rod 1D4 dropped in the reactor core. At 0507 hours, the Operating crew entered the abnormal procedure for a dropped control rod (AA/-). Based upon the dropped rod causing the reactor to go subcritical, the abnormal procedure directs all rods to be inserted. The reactor was manually tripped as directed by the Control Room Supervisor. At the time the control rod dropped, low power physics testing was being performed for the return of the Unit to power operation following the 13th refueling outage. The unit was stabilized in Mode 3 with the reactor at normal operating temperature and pressure with core cooling provided by the auxiliary feedwater system and main steam dumps to the condenser.

This report is being made in accordance with 10CFR50.73(a)(2)(iv)(A), "any event or condition that resulted in manual or automatic actuation of any of the systems listed in paragraph (a)(2)(iv)(B)."

CAUSE OF OCCURRENCE

The cause of the dropped rod and the subsequent manual reactor trip is attributed to a blown fuse (10 amp) for control rod 1D4. The fuse blew as a result of infant mortality. During the 13th refueling outage, preventive maintenance activities were performed to replace the rod control system power cabinet fuses for stationary gripper, movable gripper and lift coils. When the rod control system was energized following the preventive maintenance, several fuses blew and were replaced.

In addition to the infant mortality discussed above, the present 10 amp fuses in the rod control system have minimal margin to prevent failure of the fuse during maximum peak current periods based on a study performed by the vendor of the rod control system. This minimal margin may have also contributed to the fuse blowing.

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

PRIOR SIMILAR OCCURRENCES

A review of LERs for Salem and Hope Creek for the previous two years did not identify any reportable events as the result of a fuse failure. A further review of dropped control rod events revealed two reportable occurrences LER 272/99-004-00 and 272/00-003-00.

In LER 272/99-004-00, Salem Unit 1 experienced an automatic reactor trip due to a dropped control rod. In this event, the fuses blew for control rod 1A3 as a result of a containment penetration conductor failure. The fuses appropriately opened to protect the containment penetration from being damaged.

In LER 272/00-003-00, Salem Unit 1 experienced an automatic reactor trip when insufficient voltage was provided to the stationary gripper coils for the Group B control rods causing the Group B rods to drop into the core. The cause was attributed to a defective voltage regulation card.

Although these events resulted in the dropping of control rods and an automatic reactor trip, the corrective actions associated with these events would not have prevented this current event from occurring.

SAFETY CONSEQUENCES

There were no safety consequences associated with the manual reactor trip. Operators appropriately responded to the dropped rod in accordance with procedures and manually tripped the reactor as directed by the Control Room Supervisor. Plant response to the manual reactor trip was normal. All safety systems operated as required.

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.

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CORRECTIVE ACTIONS:

1. In accordance with engineering evaluations and vendor recommendations, a design change was implemented to replace the existing 10 amp fuses with higher amperage fuses in the Unit 2 rod control system. The rods were tested and operated satisfactorily.
2. A similar design change to replace the existing 10 amp fuses with higher amperage fuses is being developed for the Unit 1 rod control system.

COMMITMENTS

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