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

LER 311/01-006-00
SALEM GENERATING STATION - UNIT 2
FACILITY OPERATING LICENSE NO. DPR-75
DOCKET NO. 50-311

Gentlemen:

This Licensee Event Report entitled "ECCS Leakage Outside Containment Exceeded Dose Analysis Limits" is being submitted pursuant to the requirements of 10CFR50.73 (a)(2)(v)(C) & (a)(2)(v)(D).

Sincerely,

D.F. Garchow

D. F. Garchow
Vice President -
Operations

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

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 2	05000311	1 OF 5

4. TITLE
ECCS LEAKAGE OUTSIDE CONTAINMENT EXCEEDED DOSE ANALYSIS LIMITS

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 05000
09	06	01	01	– 006 –	00	11	01	01	FACILITY NAME	DOCKET NUMBER 05000

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)	X	50.73(a)(2)(v)(C)		
	20.2203(a)(2)(iv)		50.73(a)(2)(i)(A)	X	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)		0.73(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 September 6, 2001, elevated temperatures were identified on the 2CV49 valve leak off line. The 2CV49 valve is a manual valve on the suction of the 22 Charging/Safety Injection pump (high-head safety injection). At this same time the Reactor Coolant System (RCS) unidentified leakage rate had increased to 0.26 gpm. The packing for valve 2CV49 was adjusted on September 6, and an RCS leakage was measured. The RCS unidentified leakage rate following the packing adjustment to 2CV49 was determined to be 0.15 gpm. Equating the amount of unidentified RCS leakage before and after the packing was adjusted on the 2CV49 valve would conservatively quantify the leakage through the 2CV49 packing at 0.11 gpm. The leakage from 2CV49 is classified as unfiltered leakage since it is not directed through a charcoal filter. The 0.11 gpm of unfiltered leakage exceeds the LOCA dose analysis assumption for emergency core cooling system leakage outside containment.

The cause of this event is attributed to having the wrong packing configuration and torque requirements specified on the packing data sheet for valve 2CV49. Although the torque on the packing was adjusted and the leakage through the valve diminished, a work order was initiated to repack the 2CV49 valve to the correct packing configuration. The packing data sheet for 2CV49 will also be corrected to reflect the correct packing configuration and torque requirements. This event is reportable in accordance with 10CFR50.73(a)(2)(v), "any event or condition that could have prevented the fulfillment of the safety function of structures or system that are needed to:....(C) control the release of radioactive material; or (D) mitigate the consequences of an accident."

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

PLANT AND SYSTEM IDENTIFICATION

Westinghouse – Pressurized Water Reactor

Chemical and Volume Control System (CVCS) {CB/-}

* 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 1 at 100% power during this event.

DESCRIPTION OF OCCURRENCE

On September 6, 2001, elevated temperatures were identified on the 2CV49 valve leak off line. The 2CV49 valve is a manual valve on the suction of the 22 Charging/Safety Injection pump (high-head safety injection) {CB/-}. At this same time the Reactor Coolant System (RCS) unidentified leakage rate had increased to 0.26 gpm. The packing for valve 2CV49 was adjusted on September 6, and RCS leakage was measured. The RCS unidentified leakage rate following the packing adjustment to 2CV49 was determined to be 0.15 gpm. Equating the amount of unidentified RCS leakage before and after the packing was adjusted on the 2CV49 valve would conservatively quantify the leakage through the 2CV49 packing at 0.11 gpm.

The leakage from 2CV49 is classified as unfiltered emergency core cooling system (ECCS) leakage outside the containment since it is not directed through the charcoal filter of the Auxiliary Building Ventilation (ABV) system. The 0.11 gpm of unfiltered leakage exceeds the LOCA dose analysis assumption. The dose analysis assumption for ECCS leakage outside containment ensures that following a LOCA, the radioactive releases will remain within the requirements of 10CFR100 for offiste releases and 10CFR50 Appendix A General Design Criterion 19 (GDC-19) for exposure to Control Room Operators. There is sufficient margin between the current dose analysis and 10CFR100 limits for offsite releases such that the above increase in ECCS leakage would not exceed 10CFR100 limits. However, in regard to the GDC-19 limits for exposure to the Control Room Operators, the increase in ECCS leakage would exceed the limits of GDC-19. Therefore this event is reportable in accordance with 10CFR50.73(a)(2)(v), "any event or condition that could have prevented the fulfillment of the safety function of structures or system that are needed to:....(C) control the release of radioactive material; or (D) mitigate the consequences of an accident."

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

ANALYSIS OF OCCURRENCE

The concern associated with the amount of leakage through the packing of valve 2CV49 is a concern in the event of a Loss of Coolant Accident (LOCA) with the emergency core cooling system (ECCS) taking suction from the containment sump. Following a LOCA, radioactive iodine will be entrained in the fluid entering the sump. When the ECCS is aligned to take suction from the containment sump, the radioactivity is circulated through the ECCS. Leakage from the components in the ECCS that are outside of the containment building will then cause radioactive iodine to leak from the system into the Auxiliary Building. The leakage from the ECCS components outside containment then enters the Auxiliary Building Ventilation (ABV) System. Based on the design of the ABV system, the LOCA dose analysis only assumes that a portion of the radioactive iodine that leaks from the ECCS is routed through a charcoal filter and the remainder of the leakage leaves the Auxiliary Building unfiltered. The LOCA dose analysis assumes a total of 7600 cc/hr (~ 0.033 gpm) of leakage from the ECCS outside containment and that 35% of this leakage is unfiltered.

Equating the amount of unidentified RCS leakage before and after the packing was adjusted on the 2CV49 valve would conservatively quantify the leakage through the 2CV49 packing at 0.11 gpm. The leakage from 2CV49 is classified as unfiltered leakage since it is not directed through the charcoal filter. The 0.11 gpm of unfiltered leakage exceeds the LOCA dose analysis assumption. The dose analysis assumption for ECCS leakage outside containment ensures that following a LOCA, the radioactive releases will remain within the requirements of 10CFR100 for offiste releases and 10CFR50 Appendix A General Design Criterion 19 (GDC-19) for exposure to Control Room Operators. There is sufficient margin between the current dose analysis and 10CFR100 limits for offsite releases such that the above increase in ECCS leakage would not exceed 10CFR100 limits. However, in regard to the GDC-19 limits for exposure to the Control Room Operators, the increase in ECCS leakage would exceed the limits of GDC-19. Therefore this event is reportable in accordance with 10CFR50.73(a)(2)(v), "any event or condition that could have prevented the fulfillment of the safety function of structures or system that are needed to:....(C) control the release of radioactive material; or (D) mitigate the consequences of an accident."

CAUSE OF OCCURRENCE

The cause of this event is attributed to having the wrong packing configuration and torque requirements specified on the packing data sheet for valve 2CV49. Based on a review of past work performed on valve 2CV49, the last time the valve was repacked was in 1991 using the incorrect packing data sheet. Use of the wrong packing and torque requirements will result in leakage through the packing.

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

PRIOR SIMILAR OCCURRENCES

A review of LERs for Salem Unit 1, Salem Unit 2 and Hope Creek over the past two years identified no instances in which ECCS leakage outside containment exceeded dose analysis limits due to valve packing leaks.

SAFETY CONSEQUENCES AND IMPLICATIONS

As stated in the "Analysis of Occurrence" section of this LER, the additional leakage from valve 2CV49 would not have exceeded the limits of 10CFR100 for offsite releases. However, the limits of GDC-19 for exposure of the Control Room Operators would have been exceeded as determined by the LOCA dose analysis had a LOCA occurred while this leakage existed. The LOCA dose analysis calculation is a conservative model used to determine the effect of the radioactive release to the control room operators. This model does not assume any compensatory measures are taken by the operators to reduce their exposure to the radioactive release beyond the control room emergency air conditioning system aligning to its post-accident configuration. In the event the ECCS leakage exceeded the limit in the dose analysis, control room operators can don self-contained breathing apparatuses (SCBAs) to minimize their radiation exposure. If SCBAs are worn the dose to the control room operators is reduced to essentially zero.

In accordance with Technical Specification 6.8.4.a, "Primary Coolant Sources Outside Containment," Salem station has a program to monitor leakage outside the containment and take action to reduce the leakage within the assumption of the LOCA dose analysis. Through implementation of this program the increase in RCS unidentified leakage was determined to be outside containment and actions were expeditiously taken to minimize the leakage. Based on the above, there was no impact to the health and safety of the public.

A review of this condition determined that a Safety System Functional Failure (SSFF) has occurred as defined in Nuclear Energy Institute (NEI) 99-02.

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

CORRECTIVE ACTIONS:

1. The valve packing data sheet for 2CV49 valve will be revised to reflect the correct packing configuration and torque requirements. This action is being tracked in accordance with PSEG Nuclear's corrective action program.

2. Although the torque on the packing was adjusted and the leakage through the valve diminished, a work order was initiated to repack the 2CV49 valve to the correct packing configuration.

3. The extent of condition of the incorrect packing configuration will be evaluated 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.