



OCT 06 2005

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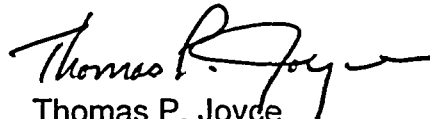
U. S. Nuclear Regulatory Commission  
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Washington, DC 20555

LER 272/04-005-01  
SALEM - UNIT 1  
FACILITY OPERATING LICENSE NO. DPR-70  
DOCKET NO. 50-272

The attached Licensee Event Report (LER) supplement, "ECCS Leakage Outside Containment Exceeds Dose Analysis Limits (11 RHR Heat Exchanger)," is being submitted to incorporate the results of the root cause analysis that was in progress at the time of the original submittal. The original LER was submitted on February 3, 2005.

The attached supplement Licensee Event Report contains no commitments.

Sincerely,

  
Thomas P. Joyce  
Site Vice President - Salem

Attachment

/EHV

C      Distribution  
         LER File 3.7

JE22

## LICENSEE EVENT REPORT (LER)

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

Estimated burden per response to comply with this mandatory 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 and FOIA/Privacy Service Branch (T-5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to [infocollects@nrc.gov](mailto:infocollects@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 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 collection.

1. FACILITY NAME <b>Salem Generating Station Unit 1</b>	2. DOCKET NUMBER <b>05000272</b>	3. PAGE <b>1 OF 5</b>
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4. TITLE <b>ECCS Leakage Outside Containment Exceeds Dose Analysis Limits (11 RHR Heat Exchanger)</b>
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5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
12	05	2004	2004	- 005 -	01	10	06	2005		

9. OPERATING MODE  4	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR§: (Check all that apply)									
10. POWER LEVEL  0%	<input type="checkbox"/> 20.2201(b)	<input type="checkbox"/> 20.2203(a)(3)(i)	<input type="checkbox"/> 50.73(a)(2)(i)(C)	<input type="checkbox"/> 50.73(a)(2)(vii)						
	<input type="checkbox"/> 20.2201(d)	<input type="checkbox"/> 20.2203(a)(3)(ii)	<input type="checkbox"/> 50.73(a)(2)(ii)(A)	<input type="checkbox"/> 50.73(a)(2)(viii)(A)						
	<input type="checkbox"/> 20.2203(a)(1)	<input type="checkbox"/> 20.2203(a)(4)	<input type="checkbox"/> 50.73(a)(2)(ii)(B)	<input type="checkbox"/> 50.73(a)(2)(viii)(B)						
	<input type="checkbox"/> 20.2203(a)(2)(i)	<input type="checkbox"/> 50.36(c)(1)(i)(A)	<input type="checkbox"/> 50.73(a)(2)(iii)	<input type="checkbox"/> 50.73(a)(2)(ix)(A)						
	<input type="checkbox"/> 20.2203(a)(2)(ii)	<input type="checkbox"/> 50.36(c)(1)(ii)(A)	<input type="checkbox"/> 50.73(a)(2)(iv)(A)	<input type="checkbox"/> 50.73(a)(2)(x)						
	<input type="checkbox"/> 20.2203(a)(2)(iii)	<input type="checkbox"/> 50.36(c)(2)	<input type="checkbox"/> 50.73(a)(2)(v)(A)	<input type="checkbox"/> 73.71(a)(4)						
<input type="checkbox"/> 20.2203(a)(2)(iv)	<input type="checkbox"/> 50.46(a)(3)(ii)	<input type="checkbox"/> 50.73(a)(2)(v)(B)	<input type="checkbox"/> 73.71(a)(5)							
<input type="checkbox"/> 20.2203(a)(2)(v)	<input type="checkbox"/> 50.73(a)(2)(i)(A)	<input checked="" type="checkbox"/> 50.73(a)(2)(v)(C)	<input type="checkbox"/> OTHER							
<input type="checkbox"/> 20.2203(a)(2)(vi)	<input type="checkbox"/> 50.73(a)(2)(i)(B)	<input type="checkbox"/> 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A							

12. LICENSEE CONTACT FOR THIS LER	
FACILITY NAME <b>E. H. Villar, Licensing Engineer</b>	TELEPHONE NUMBER (Include Area Code) <b>856-339-5456</b>

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
A	BP	HX		No					

14. SUPPLEMENTAL REPORT EXPECTED					15. EXPECTED SUBMISSION DATE		
<input type="checkbox"/> YES (If yes, complete 15. EXPECTED SUBMISSION DATE)					<input checked="" type="checkbox"/> NO		
					MONTH	DAY	YEAR

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

On December 5, 2004, the 11 Residual Heat Removal (RHR) loop was placed in service in accordance with plant operating procedures to support forced outage activities. Following the start of the 11 RHR pump, an equipment operator (non licensed operator) performed a walk down of the system and verified that the 11 RHR train was operating properly with no leakage observed. Approximately one (1) hour after the initial verification, a health physics technician (non licensed personnel) reported that the 11 RHR heat exchanger (HX) had developed a leak. Maintenance and Operations personnel performed a walk down of the system and identified the leakage to be 0.50 gpm at the top of the RHR HX (top flange). The leakage was limited to a small area on the top of the heat exchanger flange.

The cause of this event was determined to be personal error. During the last Unit 1 refueling outage while preparing the work package to replace the heat exchanger gasket, the design engineer preparer and reviewer performed an Equivalent Change Package (ECP) instead of utilizing the required Design Change Package (DCP) process.

Because the identified leakage rate exceeded the assumptions made in the dose analysis calculation for emergency core cooling system (ECCS) leakage outside the containment, 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."

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

**PLANT AND SYSTEM IDENTIFICATION**

Residual Heat Removal System (RHR) {BP}  
Residual Heat Removal Heat Exchanger {BP/HX}

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

**IDENTIFICATION OF OCCURRENCE**

Event Date: December 5, 2004

Discovery Date: December 5, 2004

**CONDITIONS PRIOR TO OCCURRENCE**

Salem Unit 1 was in Mode 4 (HOT SHUTDOWN) at 0% power at the time of the event. No structures, systems or components were inoperable at the time of the occurrence that contributed to the event.

**DESCRIPTION OF OCCURRENCE**

On December 5, 2004, the 11 Residual Heat Removal (RHR) {BP} loop was placed in service in accordance with plant operating procedures to support forced outage activities due to the presence of oil in the Delaware River. The oil in the river was the result of a spill from an oil tanker located approximately 70 miles from the site and not associated with any plant activity.

Following the start of the 11 RHR pump, an equipment operator (non licensed operator) performed a walk down of the system and verified that the 11 RHR pump train was operating properly without any observable leakage. Approximately one (1) hour after the initial verification, a health physics technician (non licensed personnel) reported that the 11 RHR heat exchanger {BP/HX} had a leak. Maintenance and Operations personnel performed a walk down of the system and identified the leakage to be at the top of the residual heat removal heat exchanger (top flange). The leakage was limited to a small area on the top of the heat exchanger flange.

The leakage rate was estimated to be approximately 0.50 gpm. This leakage rate exceeded the administrative limit stated in the Updated Final Safety Analysis Report (UFSAR) Section 6.3.2.11 for emergency core cooling system (ECCS) leakage outside the containment.

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**DESCRIPTION OF OCCURRENCE (cont'd)**

The dose analysis assumption for ECCS leakage outside containment ensures that following a Loss of Coolant Accident (LOCA) the radioactive releases will remain within the requirements of 10CFR100 for offsite releases and 10CFR50 Appendix A General Design Criterion 19 (GDC-19) for exposure to Control Room Operators. The leakage was stopped at approximately 16:51 on December 5, 2004 by hot retorquing fourteen (14) nuts for seven (7) studs.

Because the identified leakage rate exceeded the assumptions made in the dose analysis calculation for emergency core cooling system (ECCS) leakage outside the containment, 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."

**CAUSE OF OCCURRENCE**

The cause of this event was determined to be personal error.

The heat exchanger gasket was replaced during the last Unit 1 refueling outage. However, as part of the root cause, it was determined that during the work package preparation to replace the gasket, the design engineer preparer and reviewer performed an Equivalent Change Package (ECP) instead of utilizing the required Design Change Package (DCP) process. A lack of knowledge (both training and expertise) of gaskets caused the preparer to rely solely on the vendor's expertise and recommendation; the vendor incorrectly informed him that this was an equivalent gasket replacement. This error directly led to:

- Misapplication of ECP Procedure. As a result, the evaluation did not account for critical loading conditions, (i.e., mission time, thermal, deadweight and seismic loads in addition to the gasket sealing loads requirements).
- The gasket selected and installed altered the flange joint design configuration of the heat exchanger, (determined to be an actual design change).

**PREVIOUS OCCURRENCES**

A review of reportable events for Salem and Hope Creek in the last two years identified two prior similar occurrences.

LER 311/01-006 issued November 1, 2001, titled "ECCS Leakage Outside Containment Exceeded Dose Analysis Limits."

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

**PREVIOUS OCCURRENCES (cont'd)**

The cause of this event was attributed to having the wrong packing configuration and torque requirements specified on the packing data sheet for the affected component (valve 2CV49).

LER 311/04-009 issued December 13, 2004, titled " ECCS Leakage Outside Containment Exceeds Dose Analysis Limits (23 Charging Pump)."

The apparent cause for this event was excessive leakage due to the failure of the 2CV64 to provide full isolation of the pump during maintenance.

The corrective actions taken were appropriate and specific for these events; but they would not have been expected to prevent this occurrence.

**SAFETY CONSEQUENCES AND IMPLICATIONS**

There was no actual safety consequences associated with this event. There was no event that would have caused the assumptions of the dose analysis to be exceeded.

The 0.50 gpm leakage 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 had a LOCA occurred while this leakage existed as determined by a review of the LOCA dose analysis.

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 could don self-contained breathing apparatuses (SCBAs) to minimize their thyroid radiation exposure. If SCBAs were worn the thyroid dose to the control room operators would be reduced to a level that is a small fraction of the GDC 19 limit. However, the whole body control room dose to the operators could be expected to exceed the GDC 19 limits by a small amount unless more realistic input assumptions relating to such factors as containment release rates and atmospheric dispersion are credited.

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 Reactor Coolant System (RCS) unidentified leakage was determined to be outside containment and actions were expeditiously taken to minimize the leakage.

## LICENSEE EVENT REPORT (LER)

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**SAFETY CONSEQUENCES AND IMPLICATIONS (cont'd)**

Based on the above, there was no impact to the health and safety of the public.

A review of this event determined that a Safety System Functional Failure (SSFF) as defined in NEI 99-02 occurred because this event could have impacted the ability of the system to control the release of radioactive material.

**CORRECTIVE ACTIONS**

1. Engineering personnel will be provided training, as appropriate, in the proper selection of gaskets, torque calculations, and hot retorque requirements.
2. Design/Procurement engineering personnel will be provided training, as appropriate, on the design modification process to ensure that the proper modification type is selected.
3. The design documentation was revised.

**COMMITMENTS**

This LER contains no Commitments.