PSEG Nuclear LLC
P.O. Box 236, Hancocks Bridge, New Jersey 08038-0236

PSEG
Nuclear LLC

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

LER 272/04-005-00
SALEM - UNIT 1
FACILITY OPERATING LICENSE NO. DPR-70
DOCKET NO. 50-272

This Licensee Event Report, "ECCS Leakage Outside Containment Exceeds Dose Analysis Limits (11 RHR Heat Exchanger)," is being submitted pursuant to the requirements of the Code of Federal Regulations 10CFR50.73(a)(2)(v).

The attached LER contains no commitments.

Sincerely,

Carl Fricker
Salem Plant Manager

Attachment

/EHV

C Distribution
LER File 3.7

IE22

NRC FORM 366 (6-2004)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB: NO. 3150-0104	EXPIRES: 06/30/2007

LICENSEE EVENT REPORT (LER)

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

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Salem Generating Station Unit 1	05000272	1 OF 4

4. TITLE
ECCS Leakage Outside Containment Exceeds Dose Analysis Limits (11 RHR Heat Exchanger)

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 -	00	02	03	2005	FACILITY NAME	DOCKET NUMBER

9. OPERATING MODE: 4

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)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☐ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☒ 50.73(a)(2)(v)(C)	☐ OTHER
☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

FACILITY NAME	TELEPHONE NUMBER (Include Area Code)
E. H. Villar, Licensing Engineer	856-339-5456

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
D	BP	HX		No					

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

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 apparent cause for the excessive leakage has been determined to be inadequate torquing of the flange. Retorquing fourteen (14) nuts for seven (7) studs stopped the leak. Appropriate maintenance procedures will be revised to provide additional torquing instructions

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

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 offiste 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 apparent cause for the excessive leakage has been determined to be inadequate torquing of the flange during the last refueling outage (March 30, through June 3, 2004). The gasket was torqued according to the vendor's instructions; however, the vendor instructions did not require hot retorquing of the flange.

A formal root cause investigation is in progress. Upon completion of the root cause this LER may be supplemented if the findings and conclusions are significantly different from what is stated herein.

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."

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.

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

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.

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. Fourteen (14) nuts for seven (7) studs were initially hot retorqued to stop the leak.

2. The appropriate maintenance procedures will be revised to provide additional torquing instructions.

COMMITMENTS

This LER contains no Commitments.