PSEG Nuclear LLC
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PSEG
Nuclear LLC

AUG 0 8 2013

10CFR50.73

LR-N13-0163

United States Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-001

Hope Creek Generating Station Unit 1
Renewed Facility Operating License No. NPF-57
Docket No. 50-354

Subject: Licensee Event Report 2013-003-00

In accordance with 10 CFR 50.73(a)(2)(ii)(A) and 10 CFR 50.73(a)(2)(i)(B), PSEG Nuclear LLC is submitting Licensee Event Report (LER) Number 2013-003-00, "Through-wall Flaw Discovered on RHR Shutdown Cooling Return Vent Line."

Should you have any questions concerning this letter, please contact Mr. Paul Bonnett at (856) 339-1923.

No regulatory commitments are contained in the LER.

Sincerely,

Eric S. Carr
Plant Manager
Hope Creek Generating Station

Attachment: Licensee Event Report 2013-003-00

cc: Mr. W. Dean, Regional Administrator – Region I
U.S. Nuclear Regulatory Commission
2100 Renaissance Blvd, Suite 100
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USNRC Senior Resident Inspector – Hope Creek (X24)

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New Jersey Department of Environmental Protection
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MC 33-01
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Hope Creek Commitment Tracking Coordinator (H02)

Corporate Commitment Tracking Coordinator (N21)

NRC FORM 366 (10-2010) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB: NO. 3150-0104 EXPIRES: 10/31/2013

Estimated burden per response to comply with this mandatory collection request: 80 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 F53), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to 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.

LICENSEE EVENT REPORT (LER)

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

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4. TITLE

Through-wall Flaw Discovered on RHR Shutdown Cooling Return Vent Line

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
									N/A	
06	13	2013	2013	- 003 -	00	08	08	2013	FACILITY NAME: N/A	DOCKET NUMBER

9. OPERATING MODE

3

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)
Paul Bonnett, Sr. Compliance Engineer	856-339-1923

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	BO	PSF		N					

14. SUPPLEMENTAL REPORT EXPECTED

☒ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)* ☐ NO

15. EXPECTED SUBMISSION DATE

MONTH	DAY	YEAR
09	06	2013

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

On June 13, 2013, Hope Creek Unit 1 was in Operational Condition (OPCON) 3 following a scram that occurred on June 12, 2013. At approximately 02:52 EDT, during the initial drywell walk down, operators observed water leaking from the 'B' Residual Heat Removal (RHR) Shutdown Cooling Return Vent Line. The source of the leak was identified as a through-wall flaw at the pipe/weld interface on the upstream side of the 'B' RHR vent line outboard isolation valve (BC-V597), which is inside the reactor coolant system pressure boundary. The estimated leakage rate through the through-wall flaw was determined to be less than one gallon per minute. The RHR vent line is 1" ASME Class 1 piping.

Corrective actions included replacing the vent line, including both inboard (BC-V589) and outboard (BC-V597) isolation valves, during the forced outage.

An apparent cause evaluation is in progress. The results of the evaluation will be published in a supplement to this LER.

This condition is reportable under 10 CFR 50.73(a)(2)(ii)(A) for a condition that resulted in a principal safety barrier being seriously degraded. Based on the visual inspection performed during the drywell walkdown, the leak existed during plant operation. The Technical Specification limits RCS pressure boundary leakage to zero; therefore, this condition is reportable under 10 CFR 50.73(a)(2)(i)(B) for a condition prohibited by Technical Specifications.

NRC FORM 366A (10-2010)

LICENSEE EVENT REPORT (LER) U.S. NUCLEAR REGULATORY COMMISSION

CONTINUATION SHEET

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NARRATIVE

PLANT AND SYSTEM IDENTIFICATION

General Electric – Boiling Water Reactor (BWR/4)
Residual Heat Removal System Pipe Fittings - {BO/PSF}* - EIIS Identifier

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

IDENTIFICATION OF EVENT

Event Date: June 13, 2013
Discovery Date: June 13, 2013

CONDITIONS PRIOR TO EVENT

Hope Creek was in Operational Condition (OPCON) 3 at 0 percent rated thermal power (RTP) when the condition was discovered. The 'C' service water system was inoperable for planned maintenance. No other structures, systems, or components were INOPERABLE at the start of this event or contributed to the event.

DESCRIPTION OF EVENT

On June 13, 2013, Hope Creek Unit 1 was in OPCON 3 following a scram that had occurred the previous day. At approximately 02:52 EDT, during the initial drywell walk down, operators observed water leaking from the 'B' Residual Heat Removal (RHR) Shutdown Cooling Return Vent Line {BO/PSF}. The source of the leak was identified as a through-wall flaw at the pipe/weld interface on the upstream side of the 'B' RHR vent line outboard isolation valve (BC-V597). The 'B' RHR vent line inboard isolation valve (BC-V589) is normally closed. The estimated leakage rate through the flaw was determined to be less than one gallon per minute. The length of the flaw was approximately 1" at the outer diameter of the weld and approximately 7/16" at the inner diameter. The width of the flaw was approximately 1/8" at the outer diameter of the pipe. The vent line is 1" ASME Class 1 piping, located downstream of the BC-HV-F050B Shutdown Cooling Return Check Valve, within the reactor coolant system (RCS) pressure boundary.

At 09:41 EDT, on June 13, 2013, Hope Creek made a 8-hour notification to the NRC under 10 CFR 50.72(b)(3)(ii)(A) for a condition that resulted in a principal safety barrier being seriously degraded (Event Number 49110). This condition is reportable under 10 CFR 50.73(a)(2)(ii)(A) for a condition that resulted in a principal safety barrier being seriously degraded. Based on the visual inspection performed during the drywell walkdown, the leak existed during plant operation. The Technical Specification limits RCS pressure boundary leakage to zero, therefore this condition is reportable under 10 CFR 50.73(a)(2)(i)(B) for a condition prohibited by Technical Specifications.

CAUSE OF EVENT

An apparent cause evaluation is in progress. The results of the evaluation will be published in a supplement to this LER.

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NARRATIVE

SAFETY CONSEQUENCES AND IMPLICATIONS

The safety consequences of this event are low. The RCS leakage resulted from a through-wall flaw on the vent line on the upstream side of the outboard isolation valve of an isolated vent line with the inboard isolation valve closed. The RCS pressure boundary leakage was contained within the drywell and did not interface with any other systems. The Technical Specification limits RCS pressure boundary leakage to zero; however, unidentified RCS pressure boundary leakage remained within Technical Specification limits.

From a qualitative perspective, there was no significant increase in core damage frequency or large early release frequency due to the RCS pressure boundary leak. The amount of leakage from the vent line was within the makeup capability of the high pressure coolant injection (HPCI) system.

The vent line, including the inboard and outboard isolation valves, was replaced on June 15, 2013, during the forced outage. As part of an extent of condition, five additional vent lines in the drywell were examined via liquid penetrant testing to look for possible defects. These vent lines were selected because they were installed during the same modification to provide additional venting capability. No defects were identified in the additional vent line inspections.

SAFETY SYSTEM FUNCTIONAL FAILURE

A review of this event determined that a Safety System Functional Failure (SSFF) as defined in NEI 99-02, "Regulatory Assessment Performance Indicator Guidelines," did not occur. This event did not prevent the ability of a system to fulfill its safety function to either shutdown the reactor, remove residual heat, control the release of radioactive material, or mitigate the consequences of an accident.

PREVIOUS EVENTS

A review of Licensee Event Reports at Hope Creek for the past three years was performed to determine if a similar event had occurred. No events involving through-wall flaws or RCS pressure boundary leakage have previously occurred.

CORRECTIVE ACTIONS

1. The vent line, including both inboard (BC-V589) and outboard (BC-V597) isolation valves, were replaced during the forced outage.

2. Additional corrective actions will be determined based on the results of the apparent cause evaluation and failure analysis of the through-wall flaw.

COMMITMENTS

This LER contains no regulatory commitments.