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

PSEG
Nuclear LLC

SEP 0 6 2013
LR-N13-0212

10CFR50.73

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 2013-003-01

Reference: PSEG Letter LR-N13-0163 dated August 8, 2013
Licensee Event Report 2013-003-00

The Reference, Hope Creek Generating Station (HCGS) Licensee Event Report (LER), reported a through-wall flaw discovered on Residual Heat Removal Shutdown Cooling Return Vent Line. The Reference stated that HCGS would supply a supplement to the LER with the results of the technical evaluation performed to determine the cause of the through-wall flaw. The results of the technical evaluation are being communicated in the LER supplement attached to this letter.

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 2013-003-01

cc: Mr. W. Dean, Regional Administrator – Region I
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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

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: N/A	DOCKET NUMBER
06	13	2013	2013	- 003 -	01	09	06	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		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 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.

The cause of the leak was determined to be a human performance deficiency in completion of work in the drywell. A failure analysis performed by an external vendor indicated that the through-wall flaw was caused by grinding.

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 Reactor Coolant System pressure boundary leakage to zero; therefore, this condition is also reportable under 10 CFR 50.73(a)(2)(i)(B) for a Condition Prohibited by Technical Specifications.

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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 pump 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) in combination with seat leakage through the 'B' RHR vent line inboard valve (BC-V589). The 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 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. Failure analysis by an external vendor determined that the flaw dimensions were 0.880" long by 0.110" wide at the pipe outer diameter. The inner diameter opening measured 0.455" long by 0.125" wide.

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 also reportable under 10 CFR 50.73(a)(2)(i)(B) for a Condition Prohibited by Technical Specifications.

The vent line assembly, including the inboard isolation valve (BC-V589) and the outboard isolation valve (BC-V597), was replaced during the forced outage in June 2013. As part of an extent of condition, five additional RHR vent line assemblies 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 on the five additional vent line assemblies. The section of the vent line containing the flaw was shipped off-site to an external vendor for failure analysis.

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NARRATIVE

CAUSE OF EVENT

An apparent cause evaluation was completed for the through-wall flaw identified in the pipe/weld interface on the upstream side of the 'B' RHR vent line outboard isolation valve (BC-V597). A failure analysis was obtained from an external vendor. The analysis indicated that the through-wall flaw was caused by grinding. The conclusion was based on the symmetry of the opening at the inner diameter of the pipe, the outer diameter-to-inner diameter flowed metal on the pipe inner diameter surface, and surface cold-working on the side walls of the flaw. No evidence was present that the flawed region was caused by in-service degradation such as flow erosion, pitting corrosion, corrosion fatigue, or mechanical fatigue.

The apparent cause of the through-wall flaw at the pipe/weld interface was determined to be a human performance deficiency in completion of work in the drywell. The BC-V597 (outboard) vent valve was last manipulated for system filling and venting during an outage in April 2006. The BC-V589 (inboard) vent valve was last manipulated for the expanded 10-year In-Service System Leakage Test of the Reactor Coolant Pressure Boundary completed November 5, 2007. During the In-Service Inspection test, BC-V589 was opened and the vessel was pressurized to test the integrity of the piping to the outboard valve, BC-V597. The test was completed with satisfactory results; no leakage from the area was identified. The vessel hydro test in November 2007 would have shown a leak had the flaw in the pipe been present at that time. Although several work orders have been completed involving cutting on small bore piping in the drywell since November 2007, no work orders were identified where the BC-V597 valve or its connecting welds were specifically included as part of the work package.

The contributing cause to this event was leak-by through the inboard vent valve, BC-V589. The failure analysis report noted evidence of corrosion around the opening of the through-wall flaw. The inspection of the valve plug and seating surfaces of the BC-V589 identified evidence of steam cutting and erosion on the valve plug and seat. When wiped clean, the valve plug showed three areas where the seat had eroded away while other areas remained intact. Based on this evidence, it appears that some debris had been caught between the plug and seat, allowing corrosion in the seating area to occur and steam to pass through.

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 limit for RCS pressure boundary leakage is zero; however, RCS unidentified leakage remained within the Technical Specification limits.

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.

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NARRATIVE

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 events 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 occurred within the previous three years.

CORRECTIVE ACTIONS

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

2. Visually inspect, during the refueling outage, the components associated with work orders completed on Class 1 or Class 2 small bore piping in the drywell between 11/5/2007 and 5/7/2012 to ensure no similar condition is present and that no conditions adverse to quality exist.

COMMITMENTS

This LER contains no regulatory commitments.