PSEG Nuclear LLC
P. O. Box 236, Hancocks Bridge, NJ 08038

PSEG
Nuclear LLC

DEC 2 1 2010

LR-N10-0436 **10 CFR 50.73**

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

Hope Creek Generating Station Unit 1
Facility Operating License Number NPF-57
Docket Number 50-354

Subject: Licensee Event Report 2010-003

In accordance with 10 CFR 50.73(a)(2)(i)(B), PSEG Nuclear LLC is submitting Licensee Event Report (LER) Number 2010-003.

Should you have any questions concerning this letter, please contact Mr. Philip J. Duca at (856) 339-1640.

No regulatory commitments are contained in the LER.

Sincerely,

Lawrence M. Wagner
Plant Manager
Hope Creek Generating Station

Attachment: Licensee Event Report 2010-003

IE22
NRR

cc: Mr. W. Dean, Administrator – Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Mr. R. Ennis, Project Manager Salem and Hope Creek
U.S. Nuclear Regulatory Commission
One White Flint North
Mail Stop 08 B1A
11555 Rockville Pike
Rockville, MD 20852

USNRC Senior Resident Inspector – Hope Creek (X24)

P. Mulligan, Manager IV
Bureau of Nuclear Engineering
PO Box 415
Trenton, NJ 08625

Hope Creek Commitment Tracking Coordinator (H02)

INPO – LEREvents@INPO.org

NRC FORM 366 (10-2010)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB: NO. 3150-0104 EXPIRES: 10/31/2013
LICENSEE EVENT REPORT (LER)		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 Section (T-5 F53), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects.resource@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	2. DOCKET NUMBER	3. PAGE
Hope Creek Generating Station	05000 354	1 of 4

4. TITLE
RHR Shutdown Cooling Suction Relief Valve Missed Surveillance.

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
11	01	10	2010	0 0 3	00	12	21	2010		DOCKET NUMBER

9. OPERATING MODE: 5

10. POWER LEVEL: 000

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)
Philip J. Duca, Compliance Engineer	(856) 339-1640

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	RV	C710	Y					

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)*

In December 2009 a self assessment discovered a technical error in the testing of 1BCPSV-4425, the Residual Heat Removal (RHR) Shutdown Cooling (SDC) suction relief valve. 1BCPSV-4425 is an ASME Class One (1) valve, but had been improperly grouped with Class Two (2) and Three (3) valves. As a result 1BCPSV-4425 had been tested on a ten year, instead of a five year interval. The valve was last tested on October 25, 2007. Since there is no other Class One (1) valve of this type, it would be in a group by itself and therefore required to be tested every 24 months. As such, this discovery constituted a missed surveillance for 1BCPSV-4425. This required application of Technical Specification (TS) 4.0.3.

A risk evaluation was performed in accordance with TS 4.0.3 on December 17, 2009 to justify delay of the test until the Fall 2010 refueling outage. The evaluation concluded there was no significant increase in risk as a result of the delay in performance of the missed surveillance. The testing performed on November 1, 2010 was unsatisfactory as the as-found setpoint was outside the acceptable range. Therefore, this failed late surveillance is reportable per 10 CFR 50.73(a)(2)(i)(B).

NARRATIVE

PLANT AND SYSTEM IDENTIFICATION

General Electric – Boiling Water Reactor (BWR/4)
RHR/LPCI System (BWR) - EIIS Identifier {BO}*
Relief Valve - EIIS Identifier {RV} *

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

IDENTIFICATION OF OCCURRENCE

Event Date: November 1, 2010

Discovery Date: November 1, 2010

CONDITIONS PRIOR TO OCCURRENCE

Hope Creek was in Operational Condition Five (OPCON 5) for the sixteenth refueling outage (H1R16). No structures, systems or components were inoperable at the time of discovery that contributed to the event.

DESCRIPTION OF OCCURRENCE

In December 2009 a Hope Creek self assessment included a technical review of all components in the In-service Test (IST) program to ensure that the testing being performed matched the ASME Code requirements based on the classification of each component. This review discovered a technical error in the testing being performed on 1BCPSV-4425 {RV}, Residual Heat Removal (RHR) {BO} Shutdown Cooling (SDC) suction relief valve.

1BCPSV-4425 is an ASME Class One (1) valve, but had been improperly grouped with 10 RHR system Class Two (2) and Three (3) valves. ASME OM Code, Mandatory Appendix I, Section I-1320 specifies that Class 1 valves be tested on a maximum interval of five years, with a minimum of 20% of each valve group tested in any 24 month period. Section I-1350 governs testing frequency for Class Two (2) and Three (3) valves and specifies a maximum testing interval of ten years, with a minimum of 20% of each valve group tested in any 48 month period. As a result of the error in grouping, 1BCPSV-4425 has been, throughout the life of the plant, tested on a ten year, instead of a five year interval.

1BCPSV-4425 was last tested on October 25, 2007. Since there is no other ASME Class One (1) valve of this type, it would be in a group all by itself and therefore required to be tested every 24 months. As such, this discovery constituted a missed surveillance for 1BCPSV-4425 and required application of Technical Specification (TS) 4.0.3.

A risk evaluation was performed in accordance with TS 4.0.3 on December 17, 2009 to justify delay of the

NARRATIVE

DESCRIPTION OF OCCURRENCE (cont'd)

test until the Fall 2010 refueling outage. A bounding evaluation was performed in accordance with the procedure for risk assessments of missed or deficient surveillances. The evaluation for the TS surveillance for 1BCPSV-4425 Shutdown Cooling Suction line safety relief valve function supported a maximum deferral time equal to the normal surveillance interval. The evaluation concluded there was no significant increase in risk as a result of the missed surveillance. The surveillance test was performed during the Fall 2010 refueling outage (H1R16). The test results were unsatisfactory. The setpoint for the valve is 1250 psig. The as-found setpoint was 1346.3 psig which is outside the ± 3% acceptable range [Reference – ASME OM Code-2001, Mandatory Appendix I (Inservice Testing of Pressure Relief Devices in Light-Water Reactor Nuclear Power Plants), Section I-1300 Guiding Principles, Subsection I-1320 Test Frequencies, Class1 Pressure Relief Valves, Paragraph (c) Requirements for Testing Additional Valves, Subsection (1)] . Therefore, this failed late surveillance is reportable per 10 CFR 50.73(a)(2)(i)(B).

CAUSE OF OCCURRENCE

A review of available plant documents, including all applicable procedures and the original program data base, and discussions with the IST Engineer identified the apparent cause of the error in grouping as a technical rigor application deficiency that occurred at the inception of the Hope Creek relief valve program development in 1985.

The relief valve was replaced with a spare and the removed valve was disassembled. No replacement parts were required. The valve was reassembled, bench calibrated, and returned to stock as a spare. The most likely cause of the failure of the valve to actuate within the acceptable band is corrosion bonding or bridging.

PREVIOUS OCCURRENCES

A review of LERs for the three prior years at Hope Creek was performed to determine if a similar event had occurred. Two similar events were identified:(1) During the 2009 Hope Creek refueling outage when six Main Steam Safety Relief Valves (SRVs) were found out of the TS required limits of +/- 3%. This event was reported as LER 354/2009-002-00 and its supplement 354/2009-002-01(2) During the 2010 Hope Creek refueling outage when six Main Steam SRVs were found out of the TS required limits of +/- 3%. This event was reported as LER 354/2010-002.

SAFETY CONSEQUENCES AND IMPLICATIONS

1BCPSV-4425 is located on the common RHR shutdown cooling suction line. The valve is located between inboard and outboard isolation valves H1BC-1BCV-071 and H1-1BCV-164 respectively. The valve is a safety relief valve with a minimum design capacity of 0.1 gpm (ref. Specification 10855-M-141(Q), Appendix C). The valve is designed to relieve system pressure in the event of a primary containment isolation signal that would cause the referenced valves to close following the system

NARRATIVE

reaching maximum design values. Per the design piping specification the maximum pressure and temperature this piping is rated for is 1375 psig and 575 °F. During the as found testing the relief valve lifted at 1346.3 psig, lower than the maximum rating of the piping. The valve moving off of its seat would relieve enough pressure to meet the requirement of 0.1 gpm, since the full open capacity is 100 GPM at 10% accumulation. Therefore a 10% accumulation for this application is very conservative and the valve would have performed its design function upon initial opening.

A review of this event determined that a Safety System Functional Failure (SSFF) did not occur as defined in Nuclear Energy Institute (NEI) 99-02, “Regulatory Assessment Performance Indicator Guideline”.

CORRECTIVE ACTIONS

(1) Upon discovery of the improper grouping of 1BCPSV-4425, an immediate, detailed, item by item review of all relief devices in the IST program was performed individually by the IST Program Manager and the Relief Valve Program manager. This review was focused specifically on the ASME classification of the valves and whether or not any other components were not appropriately grouped per their classification. This review confirmed that 1BCPSV-4425 is the only IST relief device, other than the Main Steam Safety Relief Valves, that are Class 1 components. All other IST relief devices are Class 2 and 3 and assigned to groupings with the appropriate testing frequency based on Classification and function. (Complete)

(2) The operating experience of this event was rolled to the Programs Engineering Group by the Manager of that group as a re-enforcement for the need for thorough technical rigor application in the production and review of engineering products. (Complete)

(3) The procedure for testing of Hope Creek ASME class 1, 2, 3 safety/relief valves was revised to establish a new valve group for 1BCPSV-4425. (Complete)

(4) The Maintenance Plan for 1BCPSV-4425 was revised to 18 months not to exceed 24 months. (Complete)

(5) 1BCPSV-4425 was replaced with a spare (that was satisfactorily tested), rebuilt, bench calibrated and returned to stock. (Complete)