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PSEG
Nuclear LLC

10CFR50.73

LR-N16-0239

JAN 04 2017

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 2016-005-00, Reactor Protection System Actuation While the Reactor Was Shutdown

In accordance with the requirements of 10 CFR 50.73(a)(2)(iv)(A), PSEG Nuclear LLC is submitting the enclosed Licensee Event Report (LER) Number 2016-005-00, "Reactor Protection System Actuation While the Reactor Was Shutdown."

If you have any questions or require additional information, please contact Mr. Thomas MacEwen at (856) 339-1097.

There are no regulatory commitments contained in this letter.

Sincerely,

Edward T. Casulli
Plant Manager
Hope Creek Generating Station

ttm

Attachment: Licensee Event Report 2016-005-00

10CFR50.73

cc: Mr. Daniel Dorman, Regional Administrator – Region I, NRC
Ms. Carleen Parker, Project Manager - US NRC
Mr. Justin Hawkins, NRC Senior Resident Inspector – Hope Creek (X24)
Mr. Patrick Mulligan, Manager IV, NJBNE
Mr. Thomas MacEwen, Hope Creek Commitment Tracking Coordinator (H02)
Mr. Lee Marabella - Corporate Commitment Tracking Coordinator (N21)

NRC FORM 366 (06-2016)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB: NO. 3150-0104	EXPIRES: 10/31/2018

LICENSEE EVENT REPORT (LER)

(See Page 2 for required number of digits/characters for each block)

(See NUREG-1022, R.3 for instruction and guidance for completing this form http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1022/r3/)

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Hope Creek Generating Station	05000354	1 OF 3

4. TITLE
Reactor Protection System Actuation While the Reactor Was Shutdown

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 05000
11	5	2016	2016	- 005	- 00	1	4	2017	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*			
4 – Cold Shutdown	☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
	☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
	☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
	☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	■ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
10. POWER LEVEL	☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
0%	☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
	☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(C)	☐ 73.77(a)(1)
	☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(D)	☐ 73.77(a)(2)(i)
	☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(vii)	☐ 73.77(a)(2)(ii)
		☐ 50.73(a)(2)(i)(C)	☐ OTHER	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

LICENSEE CONTACT	TELEPHONE NUMBER *(Include Area Code)*
Thomas MacEwen, Principal Nuclear Engineer	856-339-1097

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

14. SUPPLEMENTAL REPORT EXPECTED		15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
■ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)*	☐ NO		02	15	2017

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

On November 5, 2016, at 0404, a Reactor Protection System (RPS) actuation occurred due to a valid scram discharge volume high water level signal. This actuation was the result of a Redundant Reactivity Control System (RRCS) Alternate Rod Insertion (ARI) signal that was inadvertently generated during testing. The reactor was in cold shutdown at the time of the RPS actuation, with all control rods inserted. The Reactor Coolant System (RCS) pressure was 830 psig to support excess flow check valve testing, and shutdown cooling was removed from service. When the RRCS/ARI actuated, the B reactor recirculation pump tripped as expected, and the scram air header depressurized as expected. The depressurization of the scram air header is a design feature of the ARI. The ARI signal established the control rod drive (CRD) system scram flow path. This resulted in a high water level in the scram discharge volume (SDV), an expected response. High water level in the scram discharge volume is an actuation signal for the RPS.

This is a condition reportable under 10 CFR 50.73(a)(2)(iv)(A) as an event or condition that resulted in a manual or automatic actuation of a listed system. The cause of the RRCS/ARI actuation is personnel error. A causal evaluation is in progress to determine why the personnel error occurred.

1. FACILITY NAME	2. DOCKET NUMBER	3. LER NUMBER		
		YEAR	SEQUENTIAL NUMBER	REV NO.
Hope Creek Generating Station	05000-354	2016	- 005	- 00

NARRATIVE

PLANT AND SYSTEM IDENTIFICATION

General Electric - Boiling Water Reactor (BWR/4)
Reactor Protection System - EIIS Identifier {JC}*
Redundant Reactivity Control System - EIIS Identifier {JC }*
Reactor Recirculation System – EIIS Identifier {AD}*
Control Rod Drive System - EIIS Identifier {AA}*
Reactor Water Cleanup System - EIIS Identifier {CE}*

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

IDENTIFICATION OF OCCURRENCE

Event Date: November 5, 2016
Discovery Date: November 5, 2016

CONDITIONS PRIOR TO OCCURRENCE

Hope Creek was in Operational Condition (OPCON) 4, Cold Shutdown, at 0 percent rated thermal power. No other structures, systems or components that could have contributed to the event were inoperable at the time of the event. The reactor pressure vessel (RPV) in-service pressure test had been completed and reactor vessel pressure had been reduced from 1005 psig to 830 psig. Pressure was being held at 830 psig while excess flow check valve testing was being performed. Shutdown cooling was out of service to support the RPV pressure test, and the B reactor recirculation {AD} pump was in service to provide forced circulation.

DESCRIPTION OF OCCURRENCE

On November 5, 2016 at 0404 a RRCS / ARI {JC} signal was generated while excess flow check valve testing was in progress. The RRCS/ARI signal was generated due to trip signals on reactor pressure vessel dome pressure high channel "B" (expected for testing) and RPV water level low channel "A" (unexpected for testing condition). The unexpected signal was generated during the performance of isolating transmitters during preps for excess flow check valve testing. This signal would have been reset in accordance with procedures if followed. There were two procedures being executed in parallel by technicians to perform the excess flow check valve testing. Both procedures have steps that instruct the technicians how to isolate transmitters to preclude them from causing a trip or isolation during the test. The supervisor decided to use one of the procedures to isolate the transmitters. The procedure that was not used provided direction to reset RRCS/ARI trips after the transmitters are isolated. This step is not included in both procedures, and consequently, was not performed.

Upon RRCS initiation, the B reactor recirculation pump tripped as expected. The ARI system depressurized the scram air header, establishing the CRD {AA} system scram flow path, and the SDV filled with water that was being discharged from the control rod drive mechanisms, as expected. When the water level reached the SDV high level set-point, an RPS {JC} actuation occurred. All rods were already inserted. Operators reset the RRCS and RPS signals and then lowered RPV pressure in accordance with procedures, and placed shutdown cooling in service.

All systems operated as expected following the trip of RPS.

NARRATIVE

CAUSE OF EVENT

The cause the event was personnel error associated with partial procedure use. A causal evaluation is in progress. A supplement to this LER will be submitted to provide the results of the investigation.

SAFETY CONSEQUENCES AND IMPLICATIONS

There were no adverse safety consequences as a result of this event. The RRCS/ARI and RPS actuation did not cause any required systems to become inoperable or any design limits to be exceeded. The RRCS is not required to be operable in Operational Condition 4. The SDV high level trip is not required to be operable in Operational Condition 4. All plant systems responded as designed. The reactor water cleanup (RWCU) {CE} system remained in service, providing decay heat removal, as planned, during the pressure test window.

SAFETY SYSTEM FUNCTIONAL FAILURE

A review of this condition and the associated evaluations determined that a Safety System Functional Failure (SSFF) as defined in Nuclear Energy Institute (NEI) 99-02, "Regulatory Assessment Performance Indicator Guideline," did not occur.

PREVIOUS EVENTS

A review of events for the past three years at Hope Creek was performed to determine if similar events had occurred. On September 28, 2015, a human error resulted in an RRCS/ARI actuation and subsequent RPS actuation with the reactor at 100 percent power. This was reported in LER 2015-005. That event involved a personnel error that was associated with incorrect keypad entries during testing on the RRCS system. The causal evaluation will review similarities to that event.

CORRECTIVE ACTIONS

The individual involved in the event was disqualified from performing this and similar duties pending remediation.

Other corrective actions will be determined following the completion of the causal evaluation.

COMMITMENTS

There are no regulatory commitments contained in this LER.