PSEG Nuclear LLC
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PSEG
Nuclear LLC

10CFR50.73

LR-N15-0258

JAN 0 5 2016

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 2015-005-01

In accordance with the requirements of 10 CFR 50.73(a)(2)(iv)(A), PSEG Nuclear LLC is submitting the enclosed Supplemental Licensee Event Report (LER) Number 2015-005-01, "Reactor Scram Due to Invalid RRCS Actuation."

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,

Eric S. Carr
Plant Manager
Hope Creek Generating Station

ttm

Attachment: Licensee Event Report 2015-005-01

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
Bureau of Nuclear Engineering
New Jersey Department of Environmental Protection
PO Box 420
Trenton, NJ 08625

Mr. Thomas MacEwen, Hope Creek Commitment Tracking Coordinator (H02)

Mr. Lee Marabella - Corporate Commitment Tracking Coordinator (N21)

NRC FORM 366 (11-2015) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB: NO. 3150-0104 EXPIRES: 10/31/2018

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 FOIA, Privacy and Information Collections Branch (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.

LICENSEE EVENT REPORT (LER)

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

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Hope Creek Generating Station	05000 354	1 OF 3

4. TITLE

Reactor Scram Due to Invalid RRCS Actuation

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
09	28	2015	2015	- 005 -	01	01	05	2015		05000
									FACILITY NAME	DOCKET NUMBER
										05000

9. OPERATING MODE	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*			
1	☐ 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)
100 %	☐ 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 Compliance 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				

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

On September 28, 2015, at 20:46, with the Hope Creek reactor operating at 100% power, a human error during surveillance testing resulted in the actuation of the Redundant Reactivity Control System (RRCS), and subsequently, an automatic reactor scram on a valid low water level signal. At the time of the transient, a surveillance test of division 1 of the RRCS system was in progress. The test simulates a high reactor pressure signal. Plant data show the signal was entered in both channels of division 1 of the RRCS system. The resulting system actuation caused a trip of both Reactor Recirculation Pumps, and the actuation of the Alternate Rod Insertion (ARI) function of the RRCS system. As a result of these two actuations, reactor power lowered, causing reactor water level to lower to the Reactor Protection System (RPS) trip set point of +12.5 inches. The RPS initiated an automatic reactor scram. Reactor operators recovered water level to within the desired band using the feedwater system. Reactor pressure was maintained using turbine bypass valves discharging to the main condenser.

This report is being submitted under 10 CFR 50.73(a)(2)(iv)(A), as an event or condition that resulted in the actuation of the Reactor Protection System.

NRC FORM 366A (11-2015) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB: NO. 3150-0104 EXPIRES: 10/31/2018

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 FOIA, Privacy and Information Collections Branch (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,

LICENSEE EVENT REPORT (LER) CONTINUATION SHEET

1. FACILITY NAME	2. DOCKET NUMBER	3. LER NUMBER		
		YEAR	SEQUENTIAL NUMBER	REV NO.
Hope Creek Generating Station	05000- 354	2015	- 005	- 01

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}*

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

IDENTIFICATION OF OCCURRENCE

Event Date: 09/28/15
Discovery Date: 09/28/15

CONDITIONS PRIOR TO OCCURRENCE

Hope Creek was in Operational Condition 1 at 100 percent rated thermal power (RTP). Redundant Reactivity Control System (RRCS) {JC}, Division 1, surveillance testing was in progress.

DESCRIPTION OF OCCURRRENCE

On 9/28/2015 at 20:46, a Hope Creek Instrument and Controls technician was performing a surveillance test of RRCS division 1, channel B, to simulate a high reactor pressure condition. The RRCS system is designed to detect and respond to an Anticipated Transient Without Scram (ATWS) condition. One indication of this condition is high reactor pressure, at or above 1071 psig. Under these conditions, the RRCS is designed to trip both Reactor Recirculation Pumps (RRPs) {AD} and initiate Alternate Rod Insertion (ARI). The RRPs are tripped to reduce core flow and increase the formation of core voids, thus reducing power. ARI provides an alternate path for control rod insertion by depressurizing the scram air header through valves separate from the RPS {JC} scram valves.

During the test, a keypad on the local RRCS panel is used to enter the test parameter, the test signal value and the channel being tested. The technician was expected to enter a test pressure signal of 1400 psig into the B channel of division 1. Plant data indicate the test pressure signal was also entered in channel A of division 1. With the 1400 psig test signal in both the A and B channels of logic, division 1 of the RRCS system actuated, causing RRPs to trip and ARI to begin control rod insertion by depressurizing the scram air header.

The change in reactor power caused a reactor water level transient which reached the RPS trip set-point of +12.5 inches. Although the control rods were already moving inward due to ARI actuation, the RPS functioned as designed to ensure reactor shutdown was completed via a scram signal. After the initial transient, plant operators stabilized reactor pressure and water level using turbine bypass valves and the feed water system, respectively.

CAUSE OF EVENT

The cause of this event is that the technician made an error in the performance of the surveillance test. The error was most likely caused by pressing the incorrect key on the common keyboard for the panel (placing the wrong channel in test). Based on a review of plant data (alarms and indications) and surveillance test simulation on the RRCS training simulator, it was concluded that the technician most likely recognized the unexpected conditions and attempted to correct his error. The technician did not understand that the pressure test signal had sealed in on the incorrect channel. When faced with an unexpected condition, the technician did not stop and seek supervisory guidance. When the test signal was subsequently entered into the correct channel, the RRCS system actuation resulted.

When the cause analysis determined that the cause was associated with a human error, and also determined the most probable error sequence, technician response to further questions could not be obtained, because the technician who was involved had resigned.

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NARRATIVE

SAFETY CONSEQUENCES AND IMPLICATIONS

There were no consequences to nuclear safety as a result of this event. The RRCS and RPS system operated as designed to shut down the reactor. All necessary support systems functioned as needed to support plant stabilization and recovery post transient.

SAFETY SYSTEM FUNCTIONAL FAILURE

A review of this condition 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 HCGS LERs from the past three years did not reveal any similar previous events.

CORRECTIVE ACTIONS

Following the event, the technician involved in the event was disqualified from performing any surveillance testing or other plant maintenance duties.

Other corrective actions are being tracked in the licensee's Corrective Action Program.

COMMITMENTS

This LER contains no regulatory commitments.