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

PSEG
Nuclear LLC

10CFR50.73

LR-N15-0172

AUG 26 2015

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 2015-004-01

In accordance with the requirements of 10 CFR 50.73(a)(2)(i)(B), PSEG Nuclear LLC is submitting the enclosed Supplemental Licensee Event Report (LER) Number 2015-004-01, "As-Found Values for Safety Relief Valve Lift Set Points Exceed Technical Specification Allowable Limit."

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,

(ACTING)
Edward T. Casulli

Eric S. Carr
Plant Manager
Hope Creek Generating Station

ttm

Attachment: Licensee Event Report 2015-004-01 cc: Mr. Daniel Dorman, Regional Administrator – Region I, NRC

Ms. Carleen Parker, Project Manager - US NRC

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 (02-2014) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB: NO. 3150-0104 EXPIRES: 01/31/2017

LICENSEE EVENT REPORT (LER)

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

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
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4. TITLE As-Found Values for Safety Relief Valve Lift Set Points Exceed Technical Specification Allowable Limit

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
06	02	15	2015	004	01	08	26	2015	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE: 1

10. POWER LEVEL: 100

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

LICENSEE CONTACT	TELEPHONE NUMER *(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
B	SB	RV	T020	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)*

On June 2, 2015, Hope Creek Generating Station (HCGS) received initial results of the 'as-found' setpoint testing for the safety relief valve (SRV) pilot stage assemblies. The initial results indicated that three SRV pilot stage assemblies had exceeded the lift settings prescribed in Technical Specification (TS) 3.4.2.1. The TS requires the SRV lift settings to be within +/- 3% of the nominal setpoint value. During the nineteenth refueling outage (H1R19), all fourteen SRV pilot stage assemblies were removed for testing at an offsite facility. Between June 2 and June 10, 2015, HCGS received the test results for the remainder of the SRV pilot valve assemblies. A total of ten of the fourteen SRV pilot stage assemblies experienced setpoint drift outside of the TS 3.4.2.1 specified values. All of the valves failing to meet the limits were Target Rock Model 7567F two-stage SRVs. This is a condition reportable under 10 CFR 50.73(a)(2)(i)(B) as an Operation or Condition Prohibited by Technical Specifications.

The cause of the setpoint drift for the ten SRV pilot stage assemblies is attributed to corrosion bonding between the pilot disc and seating surfaces, which is consistent with industry experience. This conclusion is based on previous cause evaluations and the repetitive nature of this condition at HCGS and within the BWR industry.

Technical evaluations performed to assess the aggregate safety significance of ten SRVs with out of tolerance initial lift setpoints concluded that this condition had no safety significance.

NARRATIVE

PLANT AND SYSTEM IDENTIFICATION
General Electric – Boiling Water Reactor (BWR/4)
Main Steam – EIIS Identifier {SB}*
Safety Relief Valves – EIIS Identifier {SB/RV}*

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

IDENTIFICATION OF OCCURRENCE
Event Date: June 2, 2015
Discovery Date: June 2, 2015

CONDITIONS PRIOR TO OCCURRENCE
When the reports of the 'as-found' results were received, Hope Creek was in Operational Condition (OPCON) 1 at approximately 100 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.

DESCRIPTION OF OCCURRRENCE

During the nineteenth refueling outage (H1R19) at Hope Creek Generating Station (HCGS), all 14 Main Steam Safety Relief Valves (SRV) pilot stage assemblies {SB/RV} were removed and tested at NWS Technologies. The SRVs are Target Rock Model 7567F two-stage SRVs. During the period from June 2, 2015 through June 10. 2015, HCGS received the results of the 'as-found' set pressure testing required by Technical Specification (TS) Surveillance Requirement (SR) 4.4.2.2. A total of ten of the 14 SRV pilot stage assemblies had setpoint drift outside of the required TS 3.4.2.1 tolerance values of +/-3% of nominal value.

The 'as-found' test results for the ten SRVs not meeting the TS requirements are as follows:

Valve ID	As Found (psig)	TS Lift Setting (psig)	Acceptable Band (psig)	% Difference Actual
F013C	1216	1130	1096.1 – 1163.9	7.61%
F013F	1240	1108	1074.8 – 1141.2	11.90%
F013G	1208	1120	1086.4 – 1153.6	7.86%
F013H	1148	1108	1074.8 – 1141.2	3.60%
F013J	1161	1120	1086.4 – 1153.6	3.66%
F013K	1161	1108	1074.8 – 1141.2	4.80%
F013 L	1165	1120	1086.4 – 1153.6	4.00%
F013 M	1207	1108	1074.8 – 1141.2	8.90%
F013P	1221	1120	1086.4 – 1153.6	9.00%
F013R	1169	1120	1086.4 – 1153.6	4.38%

NARRATIVE

Technical Specification (TS) 3.4.2.1 requires that the safety function of at least 13 of 14 SRVs be operable with a specified code safety valve function lift setting, within a tolerance of +/- 3%. Action (a) of TS 3.4.2.1 specifies "With the safety valve function of two or more of the above listed fourteen safety/relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours." Therefore, this is a condition reportable under 10 CFR 50.73(a)(2)(i)(B) as an Operation or Condition Prohibited by TS.

The extent of condition for this event is to expand the scope of the SRV Group 1 valve testing, per ASME OM Code Section I-1320 for Class 1 Pressure Relief Valves. However, since all 14 SRV pilot stage assemblies were removed and replaced with tested spares during the refueling outage (H1R19), the extent of condition scope was satisfied.

CAUSE OF EVENT

The cause of the setpoint drift for the ten SRV pilot stage assemblies is attributed to corrosion bonding between the pilot disc and seating surfaces, which is consistent with industry experience. This conclusion is based on previous cause evaluations and the repetitive nature of this condition at HCGS and within the BWR industry.

SAFETY CONSEQUENCES AND IMPLICATIONS

All 14 SRVs were operable during Cycle 19 and there were no events during that cycle that required operation of the SRVs. All SRVs lifted well below the Safety Limit, providing reasonable assurance that accident analysis conclusions would remain valid. The industry has recognized that corrosion bonding occurs during the operating cycle. Once an SRV lifts, the corrosion bond breaks and subsequent openings occur very close to the set point as demonstrated during testing.

Two technical evaluations were performed to assess the aggregate safety-significance of the 10 SRVs with out of tolerance initial lift setpoints and determine whether the condition would have had an adverse effect on the safety function of the valves or other affected system structures and components (SSCs). One technical evaluation looked at 1) the Reactor Pressure Vessel (RPV) over-pressure design function of the valves; 2) the impact of higher relief setpoints on other safety systems (i.e., HPCI, RCIC, and SLC); and 3) fuels considerations. The second technical evaluation looked at stress related issues (down-comer piping, supports, spargers, and torus loads).

The evaluations concluded that the as-found condition was bounded by margins which exist in current Hope Creek design analyses; thus, the aggregate effect of this condition has no Safety Significance. In all cases, the RCS would have remained within allowable limits, and safety-related systems relied upon during high-pressure events (HPCI, RCIC and SLC) would have functioned sufficiently in accordance with the station's design bases had an accident or limiting transient occurred during Cycle 19. Fuel limits were not adversely affected by this condition. Evaluation results of the stresses on down-comer piping, supports, spargers, and the torus loads showed satisfactory results.

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NARRATIVE

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 four years at Hope Creek was performed to determine if similar events had occurred. Similar events occurred during the 2012 (H1R17) and 2013 (H1R18) Hope Creek refueling outages when multiple SRVs were found out of the TS required limits of +/- 3%. These events were reported as LER 354/2012-004-01(six inoperable SRVs) and LER 354/2013-007-00 (five inoperable SRVs).

CORRECTIVE ACTIONS

1. All 14 SRV pilot stage assemblies were removed and replaced with pre-tested, certified spare pilot valves (H1R19).

2. Evaluate options for the replacement of the currently installed Target Rock two-stage SRVs with a design that eliminates setpoint drift events exceeding +/-3% and improve SRV reliability. The replacement schedule will be developed after a suitable valve is identified.

COMMITMENTS

There are no regulatory commitments contained in this LER.