PSEG Nuclear LLC
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PSEG
Nuclear LLC

10CFR50.73

LR-N17-0039

MAR 08 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: Supplemental Licensee Event Report (LER) Number 2016-003-01, "As-Found Values for Safety Relief Valve Lift Set Points Exceed Technical Specification Allowable Limit."

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 2016-003-01, "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,

Edward T. Casulli
Plant Manager
Hope Creek Generating Station

ttm

Attachment: Licensee Event Report 2016-003-01

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, NJBNE
Mr. Thomas MacEwen, Hope Creek Commitment Tracking Coordinator (H02)
Mr. Lee Marabella - Corporate Commitment Tracking Coordinator (N21)

10CFR50.73

(The bcc list should not be submitted as part of the DCD submittal – remove this page prior to submittal and make the bcc distribution accordingly)

bcc: President/Chief Nuclear Officer (N09)
Site Vice President – Hope Creek (H07)
Plant Manager – Hope Creek (H07)
Sr. Director – Hope Creek Operations (H08)
Director - Site Regulatory Compliance (N21)
Manager – Nuclear Shift Operations (H01)
Records Management

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/)

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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
10	22	2016	2016	- 003	- 01	3	8	2017	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE: 5 - Refuel

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)(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)
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
☐ 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
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 October 22, 2016, Hope Creek Generating Station (HCGS) received results that the 'as-found' set-point tests for safety relief valve (SRV) pilot stage assemblies had exceeded the lift setting tolerance prescribed in Technical Specification (TS) 3.4.2.1. The TS requires the SRV lift settings to be within +/- 3% of the nominal set-point value. During the twentieth refueling outage (H1R20), all fourteen SRV pilot stage assemblies were removed for testing at an offsite facility. Between October 22 and October 28, 2016, HCGS received the test results for all fourteen of the SRV pilot valve assemblies. A total of ten of the fourteen SRV pilot stage assemblies experienced set-point 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 set-point 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.

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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: October 22, 2016
Discovery Date: October 22, 2016

CONDITIONS PRIOR TO OCCURRENCE

When the reports of the 'as-found' results were received, Hope Creek was in Operational Condition (OPCON) 5, Refuel, 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.

DESCRIPTION OF OCCURRENCE

During the twentieth refueling outage (H1R20) 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 October 22 through October 28, 2016, 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 fourteen SRV pilot stage assemblies had set-point 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
F013A	1208	1130	1096.1 – 1163.9	6.9%
F013B	1173	1130	1096.1 – 1163.9	3.8%
F013D	1220	1130	1096.1 – 1163.9	8.0%
F013F	1170	1108	1074.8 – 1141.2	5.6%
F013G	1184	1120	1086.4 – 1153.6	5.7%
F013H	1151	1108	1074.8 – 1141.2	3.9%
F013 J	1173	1120	1086.4 – 1153.6	4.7%
F013 K	1146	1108	1074.8 – 1141.2	3.4%
F013P	1213	1120	1086.4 – 1153.6	8.3%
F013R	1224	1120	1086.4 – 1153.6	9.3%

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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 (H1R20), the extent of condition scope was satisfied.

CAUSE OF EVENT

The cause of the set-point 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

There were no instances during cycle 20 that resulted in any of the fourteen SRVs being declared inoperable 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 set-points and determine whether the condition would have had an adverse effect on the safety function of the valves or other affected systems, 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 set-points 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 20. 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.

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.

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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 2013 (H1R18) and 2015 (H1R19) Hope Creek refueling outages when multiple SRVs were found out of the TS required limits of +/- 3%. These events were reported as LER 354/2013-007-00 (five inoperable SRVs) and LER 354/2015-004-00 (ten inoperable SRVs).

CORRECTIVE ACTIONS

1. All 14 SRV pilot stage assemblies were removed and replaced with pre-tested, certified spare pilot valves (H1R20).

2. The station is planning the replacement of the currently installed Target Rock two-stage SRVs with three-stage SRVs that are expected to eliminate setpoint drift events exceeding +/-3% and improve SRV reliability. The replacement is expected to begin in the next planned refueling outage, H1R21, in the spring of 2018, pending resolution of open technical items with the valve manufacturer. The replacement will take place over several outages in order to replace all fourteen SRVs.

COMMITMENTS

There are no regulatory commitments contained in this LER.