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Nuclear LLC

JAN 16 2014

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LR-N14-005

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 2013-007-00

In accordance with the requirements of 10 CFR 50.73(a)(2)(i)(B), PSEG Nuclear LLC is submitting the enclosed Licensee Event Report (LER) Number 2013-007-00, "As-found Values for Safety Relief Valve Lift Setpoints Exceed Technical Specification Allowable Limit."

If you have any questions or require additional information, please contact Mr. Philip Duca at (856) 339-1640.

There are no regulatory commitments contained in this letter.

Sincerely,

Eric S. Carr
Plant Manager
Hope Creek Generating Station

Attachment: Licensee Event Report 2013-007-00

cc: W. Dean, Regional Administrator – Region I, NRC
J. Hughey, Project Manager - US NRC
NRC Senior Resident Inspector – Hope Creek (X24)
P. Mulligan, Manager, NJBNE
LER uploaded to ICES
P. Bonnett - Hope Creek Commitment Tracking Coordinator (H02)
L. Marabella - Corporate Commitment Tracking Coordinator (N21)

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	05000354	1 of 4

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
									N/A	
11	22	2013	2013	- 007 -	00	01	16	2014	N/A	DOCKET NUMBER

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

FACILITY NAME	TELEPHONE NUMBER (Include Area Code)
Philip Duca, Senior 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	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 November 22, 2013, Hope Creek Generating Station (HCGS) received results of the 'as-found' setpoint testing for the safety relief valve (SRV) pilot stage assemblies. The results indicated that five SRV pilot stage assemblies' setpoints had exceeded the lift settings prescribed in the Limiting Condition for Operation (LCO) for Technical Specification (TS) 3.4.2.1. The TS LCO requires the SRV lift settings to be within +/- 3% of the nominal setpoint value. All five of the valves failing to meet the limits were Target Rock Model 7567F two-stage SRVs. During the eighteenth refueling outage (H1R18), all 14 SRV pilot stage assemblies were removed and tested at an offsite test facility. A total of five of the 14 SRV pilot stage assemblies experienced setpoint drift outside of the TS 3.4.2.1 specified values. 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 five 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.

There was no actual safety consequence associated with this event.

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 EVENT

Event Date: November 22, 2013
Discovery Date: November 22, 2013

CONDITIONS PRIOR TO EVENT

When the report of the 'as-found' results was received, Hope Creek was in Operational Condition (OPCON) 1 at approximately 100 percent rated thermal power. No other structures, systems or components were inoperable at the time of the event. All SRV pilot stage assemblies had been replaced with pre-tested, certified spare pilot stage assemblies during the refueling outage.

DESCRIPTION OF EVENT

During the eighteenth refueling outage (H1R18) 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. On November 4, 2013, and November 22. 2013, engineering personnel 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 five 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. On November 4, 2013, HCGS received a report documenting the failure of SRV 'L.' On November 22, 2013, HCGS received a second report documenting the failures of SRVs 'A', 'D', 'F', and 'K.' The remaining nine SRVs were found to be in compliance with the TS.

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

Valve ID	As Found (psig)	TS Lift Setting (psig)	Acceptable Band (psig)	% Difference Actual
F013A	1170	1130	1096 – 1163	3.5%
F013D	1192	1130	1096 – 1163	5.5%
F013F	1178	1108	1075 – 1141	6.3%
F013K	1149	1108	1075 – 1141	3.7%
F013L	1175	1120	1087 – 1153	4.9%

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, as specified, 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.

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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 (H1R18), the extent of condition scope was satisfied.

CAUSE OF EVENT

The cause of the setpoint drift is attributed to corrosion bonding, which is consistent with prior determinations at HCGS and within the BWR industry. Corrosion bonding occurs when an oxide forms between the mating surfaces of the Pilot Disc (solid Stellite 21) and the seat in the Pilot Body (Stellite 6 overlay). This bridging oxide fractures when the pilot disc lifts. The load required to fracture this bridging oxide increases the lift point and can lead to pilot stage assemblies failing high during initial lift tests. Subsequent lifts following the initial 'as-found' lift are typically within setpoint tolerances. The five SRVs which exceeded the lift settings during the 'as-found' testing were within +/-3% tolerance for the second lift test, which confirms that corrosion bonding caused the high lift set drift.

The combination of materials used for the pilot disc and the pilot seat has been a known industry issue since the design of the Target Rock 2-stage SRV was initially installed. The oxygen content of the steam, in the pilot disc area, aggravates the natural corrosive reaction in the pilot disc seating area. Numerous industry attempts to resolve the oxide formation have failed to improve performance. A summary of the BWROG recommendations to improve SRV reliability with regard to setpoint drift was documented in NRC Regulatory Issue Summary 2000-12 dated August 7, 2000, "Resolution of Generic Safety Issue B-55, Improved Reliability of Target Rock Safety Relief Valves." The three modification options recommended were: (1) the installation of ion beam implanted platinum (IBAD Process) pilot valve discs; (2) the installation of Stellite 21 pilot valve discs; and (3) the installation of additional pressure actuation switches. Hope Creek has implemented options 1 and 2 with limited success. Option three has not been considered due to mixed industry results/performance.

Following a previous outage (H1R15), Southwest Research was contracted to metallurgically evaluate the Pilot Body and Disc from SRV-K (setpoint failure at +9.4%) using both stereomicroscopy and scanning electron microscopy (SEM) to determine if evidence of bonding between the mating surfaces of the disc and body was present. The SEM examinations of the seating area on the Pilot Disc showed clear evidence of brittle oxide fracture along the seating line. These sharp fracture lines are typically produced as a brittle oxide grown between two surfaces fractures as the surfaces are separated, leaving islands of the oxide on each surface. Spectra taken from various regions along the seat confirmed that portions of the oxide were being removed from the Pilot Disc seat, i.e., left behind on the seat face, as the disc lifted off the seat. These results confirm that an oxide had formed between the mating surfaces of the Pilot Disc and the seat in the Pilot Body and that this bridging oxide fractured when the disc lifted. The load required to fracture this bridging oxide increases the lift point and can lead to pilots failing high during lift tests.

SAFETY CONSEQUENCES AND IMPLICATIONS

There are no safety consequences associated with the 'as-found' setpoint drifts experienced during H1R18. The safety valve function of the SRVs operates to prevent the reactor coolant system from being pressurized above the Safety Limit of 1375 psig, which is 110% of the design pressure of 1250 psig, in accordance with the ASME Code. A total of 13 operable SRVs are required to limit reactor pressure to within ASME III allowable values for the worst case transient. The five SRVs that were found above the allowable +3% were below the design pressure and Safety Limit; therefore, the five SRVs were capable of fulfilling their design function and reactor vessel overpressure protection was not compromised. With regard to Emergency Core Cooling System (ECCS) performance, no SRVs are required to open, and therefore, setpoint drift is not a concern. For a small break LOCA, the Automatic

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Depressurization System (ADS) will cycle open SRVs. Two of the five SRVs ('A' and 'D') were ADS valves; however, the setpoint drift had no impact on the ADS (electronically controlled) or manual function of the valves.

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."

PREVIOUS EVENTS

A review of events for the past four years at Hope Creek was performed to determine if a similar event had occurred. Similar events occurred during the 2009 (H1R15), 2010 (H1R16), and 2012 (H1R17) Hope Creek refueling outages when multiple SRVs were found out of the TS required limits of +/- 3%. These events were reported as LER 354/2009-002-01, LER 354/2010-002-01, and LER 354/2012-004-01.

A previously completed SRV Setpoint Drift root cause evaluation documented that the pilot stage assemblies in the Target Rock 2-Stage SRV design have an industry wide chronic history of corrosion bonding leading to setpoint drifting.

CORRECTIVE ACTIONS

1. All 14 SRV pilot stage assemblies were removed and replaced with pre-tested, certified spare pilot valves (H1R18).
2. Replace the currently installed Target Rock two-stage SRVs with a design that eliminates setpoint drift events exceeding +/-3% and improve SRV reliability.

COMMITMENTS

This LER contains no commitments.