PSEG Nuclear LLC
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PSEG
Nuclear LLC

JUN 0 3 2009

LR-N09-0123 **10CFR50.73**

United States Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-001

Hope Creek Generating Station Unit 1
Facility Operating License No NPF-57
Docket No. 50-354

Subject: Licensee Event Report 2009-002

In accordance with 50.73(a)(2(i)(B), PSEG Nuclear LLC is submitting Licensee Event Report (LER) Number 2009-002.

Should you have any questions concerning this letter, please contact Mr. Timothy R. Devik at (856) 339-3108.

No regulatory commitments are contained in the LER.

Sincerely,

John F. Perry

John F. Perry
Plant Manager
Hope Creek Generating Station

Attachment: Licensee Event Report 2009-002

IE22
NRR

cc: Mr. S. Collins, Administrator – Region 1
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Mr. R. Ennis, Project Manager Salem and Hope Creek
U.S. Nuclear Regulatory Commission
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Rockville, MD 20852

USNRC Senior Resident Inspector – Hope Creek (X24)

P. Mulligan, Manager IV
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PO Box 415
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Hope Creek Commitment Tracking Coordinator

NRC FORM 366 U.S. NUCLEAR REGULATORY COMMISSION
(9-2007)

LICENSEE EVENT REPORT (LER)

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

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4. TITLE
As Found Values for Safety Relief Valve Lift Setpoints Exceed Technical Specification Allowable

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	
04	18	2009	2009	- 002 -	000	06	03	2009	FACILITY NAME N/A	DOCKET NUMBER

9. OPERATING MODE: 5

10. POWER LEVEL: 000

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)
Timothy R. Devik, Compliance Engineer	856-339-3108

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		09	01	2009

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

On April 18, 2009, PSEG received the initial results for the safety relief valve (SRV) pilot valve 'as-found' setpoint testing. The results indicated that two SRV pilot valve setpoint drifts exceeded Technical Specification (TS) allowable tolerance specified in TS 3.4.2.1. This specification requires SRV setpoint limits to be within +/- 3% of the specified value. The valves failing to meet limits were Target Rock Model 7567F two-stage SRVs. As-found testing was performed at an offsite test facility following Hope Creek fuel cycle fifteen. Additional SRVs were selected for testing, and subsequent results led to all 14 SRV pilot valves being removed as part of the expanded sample population. In all, a total of six of the fourteen SRV pilot valves experienced setpoint drift outside of the TS 3.4.2.1 limit.

The apparent cause for the SRV setpoint failures is under investigation. Immediate corrective action was to replace all of the pilot assemblies with tested and certified spare pilot assemblies. Since the number of SRVs outside the setpoint tolerance limit (six) was greater than the number of SRVs allowed to be inoperable by TS 3.4.2.1 (one), this condition is reportable under 10CFR50.73(a)(2)(i)(B) as any operation or condition prohibited by the plant Technical Specifications.

NRC FORM 366A (9-2007)	LICENSEE EVENT REPORT (LER) U.S. NUCLEAR REGULATORY COMMISSION CONTINUATION SHEET				
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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 {--/RV}*

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

IDENTIFICATION OF OCCURRENCE

Event Date: April 18, 2009
Discovery Date: April 18, 2009

CONDITIONS PRIOR TO OCCURRENCE

Hope Creek was in Operational Condition 5 for the fifteenth refueling outage. No structures, systems or components were inoperable at the time of discovery that contributed to the event.

DESCRIPTION OF OCCURRENCE

On April 18, 2009, Engineering personnel began to receive the results of the Main Steam Safety Relief Valve (SRV){SB/RV} (Target Rock Model 7567F) setpoint testing required by Technical Specification 4.4.2.2. The initial report documented the failure of SRVs C, and F to meet the TS 3.4.2.1 limit 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". At the time, Hope Creek was in OPCON 5 (refueling) with the reactor head removed and the reactor cavity flooded and connected to the spent fuel pool for refueling operations. An expanded scope of review was instituted and all fourteen SRV pilot valves were ultimately removed, tested and replaced.

SAFETY CONSEQUENCES AND IMPLICATIONS

A bounding analysis was performed and documented in NEDC-32511P, "SAFETY REVIEW FOR HOPE CREEK GENERATING STATION SAFETY/RELIEF VALVE TOLERANCE ANALYSIS." This analysis supported the increase in allowable Technical Specification (TS) setpoint drift from + 1 percent to + 3 percent. An individual SRV upper limit lift setpoint of 1250 psig with 13 SRVs available out of a total of 14 was assumed in the calculation. The calculated peak vessel pressure at the bottom of the reactor vessel was 1331 psig. This provides a margin of 44 psig to the ASME upset limit of 1375 psig.

In addition, the GE analysis also evaluated the mechanical loads that occur on the SRV discharge piping when a SRV actuates. The analysis established a maximum allowable percentage increase (MAPI) for each SRV line such that the allowable mechanical stresses on the SRV discharge piping would not be exceeded. Five of the six valves met their individual acceptance limits. The "A" SRV exceeded its limit of +3% with an as-found setpoint of 5.8%. The GE analysis of the mechanical stresses was performed for the Pacific Scientific (PSA) snubbers installed at the time the analysis was conducted. Hope Creek has since installed Lisega hydraulic snubbers in place of the PSA snubbers.

NARRATIVE

The discharge piping analysis contained in NEDC-32511P was re-assessed to ensure that the previous analysis (based on PSA snubbers) was still valid. A review and analysis of the differences between the snubbers determined that the PSA analysis was still bounding the present plant configuration. NEDC-32511P identifies a maximum increase in the nominal setpoint of "A" SRV to be 3%, without exceeding allowable stresses. The "A" SRV lifted at 5.8% above nominal setpoint. RIS 2005-20 and NRC Inspection Manual Part 9900 Technical Guidance on "Operability Determinations and functionality Assessments for Resolution of Degraded or Nonconforming Conditions Adverse to Quality or Safety" allow the use of the criteria in Appendix F of Section III of the ASME Boiler and Pressure Vessel code for operability determinations. The re-evaluation has determined that the ASME III allowable stresses on the piping system are higher than the stresses that would have been seen had the "A" SRV lifted. There is no present operability concern due to the replacement of this pilot assembly with a fully tested spare.

Based on the above, and because none of the SRVs exceeded the 1250 psig analyzed limit, there was no impact to the health and safety of the public.

The final test results for the SRVs that had setpoint drift outside the tolerance were as follows:

Valve ID	As Found (psig)	TS Setpoint (psig)	Acceptable Band (psig)	% Difference	
				Actual	Limit*
F013A	1195	1130	1096 – 1163	5.80%	3.00%
F013C	1203	1130	1096 – 1163	6.50%	21.80%
F013F	1163	1108	1075 – 1141	5.00%	5.50%
F013G	1156	1120	1087 – 1153	3.20%	8.70%
F013K	1212	1108	1075 – 1141	9.40%	22.40%
F013L	1170	1120	1087 – 1153	4.50%	16.30%

*The limit is based on the SRV discharge piping mechanical stress limit identified in Table 7-1 of GE analysis (NEDC-32511P) and is known as the "Maximum Allowable Pressure Increase" (MAPI).

A review of this event determined that a Safety System Functional Failure (SSFF) has not occurred as defined in Nuclear Energy Institute (NEI) 99-02.

CAUSE OF OCCURRENCE

The specific cause(s) of the failures has not yet been determined. The six SRV pilot assemblies will be disassembled and inspected to determine and document the cause of the failures. The results of the inspection will be evaluated and provided in a supplement to this LER.

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NARRATIVE

PREVIOUS OCCURRENCES

A review of LERs for the three prior years at Hope Creek was performed to determine if a similar event had occurred. There was a similar event during the 2006 Hope Creek refueling outage when three SRVs were found out of the TS required limits of +/- 3%. This event was reported as LER 354/06-003-00. Actions taken at that time were effective in that the following refueling outage did not have any SRV setpoint drift outside the allowable band on the SRVs tested.

CORRECTIVE ACTIONS

1. The pilot assembly of each failed SRV was replaced with a pre-tested, certified spare.

2. All 6 SRV pilot valve assemblies that failed will be disassembled and inspected to determine the cause of the setpoint drift. (Corrective Action Program (CAP) number: 70096933 / 0170)

3. All 14 SRV pilot valves will be removed, tested and replaced with pre-tested, certified spare pilot valves during the next refueling outage (RF16). (CAP number: 70096933 / 0180)

4. An equipment apparent cause evaluation is being performed to determine the cause(s) of the failures and determine any additional corrective actions. (CAP number: 70096933 / 0070)

COMMITMENTS

This LER contains no commitments.