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PSEG
Nuclear LLC

DEC 1 3 2001

LRN-01-0415

U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

LER 354/2001-007-00
HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NO. NPF-57
DOCKET NO. 50-354

Gentlemen:

This LER entitled "As Found Values for Safety Relief Valve Lift Setpoints Exceed Technical Specification Allowable Limits" is being submitted pursuant to the requirements of 10CFR50.73(a)(2)(i)(B). The attached LER contains no commitments.

Sincerely,

D. F. Garchow

D. F. Garchow
Vice President -
Operations

Attachment

/MGM

C Distribution
LER File 3.7

IE22

NRC FORM 366 (7-2001)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 7-31-2004
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)		Estimated burden per response to comply with this mandatory information collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records Management Branch (T-6 E6), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to bjs1@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 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

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As Found Values for Safety Relief Valve Lift Setpoints Exceed Technical Specification Allowable Limits

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MO	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
10	24	2001	2001 –	007 –	00	12	13	01	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE	4							
10. POWER LEVEL	0%							
	20.2201(b)		20.2203(a)(3)(ii)		50.73(a)(2)(ii)(B)		50.73(a)(2)(ix)(A)	
	20.2201(d)		20.2203(a)(4)		50.73(a)(2)(iii)		50.73(a)(2)(x)	
	20.2203(a)(1)		50.36(c)(1)(i)(A)		50.73(a)(2)(iv)(A)		73.71(a)(4)	
	20.2203(a)(2)(i)		50.36(c)(1)(ii)(A)		50.73(a)(2)(v)(A)		73.71(a)(5)	
	20.2203(a)(2)(ii)		50.36(c)(2)		50.73(a)(2)(v)(B)		OTHER Specify in Abstract below or in NRC Form 366A	
	20.2203(a)(2)(iii)		50.46(a)(3)(ii)		50.73(a)(2)(v)(C)			
	20.2203(a)(2)(iv)		50.73(a)(2)(i)(A)		50.73(a)(2)(v)(D)			
	20.2203(a)(2)(v)	X	50.73(a)(2)(i)(B)		50.73(a)(2)(vii)			
	20.2203(a)(2)(vi)		0.73(a)(2)(i)(C)		50.73(a)(2)(viii)(A)			
	20.2203(a)(3)(i)		50.73(a)(2)(ii)(A)		50.73(a)(2)(viii)(B)			

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER (Include Area Code)
Michael G. Mosier, Senior Licensing Engineer	856-339-5434

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 EXPECTED SUBMISSION DATE)	X	NO				

On October 24, 2001 Hope Creek Engineering personnel received the initial results of the Target Rock Model 7567F Safety Relief Valve (SRV) setpoint testing required by Technical Specification 4.4.2.2. This testing revealed that following Hope Creek Cycle 10, three of the fourteen SRVs experienced setpoint drift outside of the Technical Specification 3.4.2.1 limit of +/- 3%. The apparent cause for all three valve failures is sticking of the pilot disc. The valves were replaced with tested and certified spare valves. Since the number of SRVs outside of the setpoint tolerance limit (three) was greater than the number of SRVs (one) allowed to be inoperable by Technical Specification 3.4.2.1, this condition was determined to be reportable under 10CFR50.73(a)(2)(i)(B), as any operation or condition prohibited by the plant Technical Specifications.

These three valves will be disassembled and inspected to document the cause of the failure. In addition, since this failure mechanism could be present in all the valves, PSEG Nuclear LLC, will test all fourteen pilot valves at the next refueling outage.

TEXT *(If more space is required, use additional copies of NRC Form 366A)* **(17)**

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/CC}

CONDITIONS PRIOR TO OCCURRENCE

The plant was in the shutdown condition for Hope Creek's tenth refueling outage (RF10). No structures, systems, or components were inoperable at the time of discovery that contributed to the event.

DESCRIPTION OF OCCURRENCE

On October 24, 2001 Hope Creek Engineering personnel received the initial results of the Target Rock Model 7567F Safety Relief Valve (SRV) setpoint testing required by Technical Specification 4.4.2.2. This testing revealed that following Hope Creek Cycle 10, three of the fourteen SRVs experienced setpoint drift outside of the Technical Specification 3.4.2.1 limit of +/- 3%.

SRV's With Out-of-Tolerance Drift

Valve Id	As found (psig)	TS Setpoint (psig)	Acceptable band (psig)	% Difference
F013P	**1216**	**1120**	**1087 – 1153**	**8.6**
F013H	**1169**	**1108**	**1075 - 1141**	**5.5**
F013D	**1182**	**1130**	**1096 - 1163**	**4.6**

Since the number of SRVs outside of the setpoint tolerance limit (three) was greater than the number of SRVs (one) allowed to be inoperable by Technical Specification 3.4.2.1, this condition was determined to be reportable under 10CFR50.73(a)(2)(i)(B), as any operation or condition prohibited by the plant Technical Specifications.

TEXT *(If more space is required, use additional copies of NRC Form 366A)* **(17)**

CAUSE OF OCCURRENCE

The apparent cause for all three valve failures is sticking of the pilot disc. The industry recommended repair was to coat the pilot disc with a thin layer of platinum using an ion beam implantation process. PSEG has continued to experience some failures on these SRVs even with the industry recommended coating installed on the pilot disc. For two of the three pilot valves there is no recent history for coating of this pilot assembly at this time. The third pilot valve was coated prior to cycle 10. The ion implantation process appears to not be entirely effective at eliminating this phenomenon. PSEG will continue to monitor the performance of the coating on all applicable pilot discs.

PRIOR SIMILAR OCCURRENCES

LER 354/99-003, and LER 354/00-003, reported events where SRV setpoint drift exceeded the Technical Specification allowable limits during previous operating cycles.

LER 354/00-003 stated that corrective actions as a result of a NUPIC audit of Target Rock field services would be monitored for effectiveness. On February 6, 2001 a follow-up audit closed all outstanding corrective actions. There program is now in compliance with applicable sections of 10CFR50 Appendix B. The corrective actions have been implemented in an effective manner, which provides for a satisfactory level of confidence that the resulting program improvements will be effective.

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. A single upper limit setpoint of 1250 psig and 13 SRV's 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 psi to the ASME upset limit of 1375 psig. In addition, loads on SRV piping were reanalyzed. The analysis established an allowable percentage increase for each SRV line such that the allowable stresses would not be exceeded. None of the three valves exceeded their individual limits. Based upon this analysis, there were no safety consequences or implications involved as a result of these valves exceeding the allowable tolerance. Therefore, the public health and safety was not affected.

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

FACILITY NAME (1)	DOCKET (2) NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
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TEXT *(If more space is required, use additional copies of NRC Form 366A)* **(17)**

CORRECTIVE ACTIONS:

1. All valves including the three failed valves were removed from the plant and replaced with tested and certified spare or re-certified valves during RF10.
2. The failed valves will be dismantled, inspected, and refurbished prior to their next use.
3. PSEG will continue to monitor the performance of the coating on all applicable pilot discs.
4. PSEG Nuclear LLC, will test all fourteen pilot valves at the next refueling outage.

COMMITMENTS

The corrective actions cited in this LER are voluntary enhancements and do not constitute commitments.