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DEC 0 7 2001

PSEG
Nuclear LLC

LRN-01-0394

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

Gentlemen:

LER 354/2001-004-00
HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NO. NPF-57
DOCKET NO. 50-354

Gentlemen:

This Licensee Event Report entitled "Reactor Building Differential Pressure Controller Incorrectly Set" is being submitted pursuant to the requirements of 10CFR50.73(a)(2)(i)(B). The attached LER contains no commitments.

Sincerely,

for

D. F. Garchow
Vice President - Operations

Attachment

/PRD

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

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 information collection.

LICENSEE EVENT REPORT (LER)

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

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Reactor Building Differential Pressure Controller Incorrectly Set

5. EVENT DATE MO	DAY	YEAR	6. LER NUMBER YEAR	SEQUENTIAL NUMBER	REV NO	7. REPORT DATE MO	DAY	YEAR	8. OTHER FACILITIES INVOLVED FACILITY NAME	DOCKET NUMBER
10	08	2001	2001	– 004 –	00	12	07	2001		05000
										05000

9. OPERATING MODE: 1

10. POWER LEVEL: 95

	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)		50.73(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)
Paul Duke, Licensing Engineer	856-339-1466

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

14. SUPPLEMENTAL REPORT EXPECTED				15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE).	X	NO				

On October 8, 2001, reactor building pressure failed to meet the acceptance criteria during the reactor building integrity functional test. The acceptance criteria were selected to ensure the minimum pressure difference across all locations of the reactor building wall will be greater than or equal to 0.25 inches of vacuum water gauge under all postulated environmental conditions. Both reactor building differential pressure controllers were found to be set incorrectly to 0.50 inches of vacuum water gauge. Per the system operating procedure, they should be set to 0.55 inches of vacuum water gauge. The controllers were reset to the correct setpoint and the reactor building integrity functional test was completed satisfactorily. Based on a review of maintenance records, PSEG Nuclear believes the reactor building differential pressure controllers were set incorrectly during a maintenance or surveillance testing activity performed at some time after the previous reactor building integrity functional test completed on April 21, 2000. There were no safety consequences associated with this event. Under postulated worst case temperature differences, all primary containment leakage would still exhaust through the filtered FRVS pathway. There was no impact to the public health and safety.

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

PLANT AND SYSTEM IDENTIFICATION

General Electric – Boiling Water Reactor (BWR/4)
Filtration Recirculation and Ventilation System {BH}*
Secondary Containment {NG}*

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

IDENTIFICATION OF OCCURRENCE

Event Date: October 8, 2001
Discovery Date: October 8, 2001

CONDITIONS PRIOR TO OCCURRENCE

The plant was in OPERATIONAL CONDITION 1 (POWER OPERATION). No other structures, systems or components were inoperable at the start of this event that contributed to the event.

DESCRIPTION OF OCCURRENCE

During the reactor building integrity functional test on October 8, 2001, reactor building pressure fluctuated between 0.43 and 0.50 inches of vacuum water gauge. The purpose of the test is to demonstrate the filtration, recirculation and ventilation system (FRVS) will draw down the secondary containment to greater than or equal to 0.25 inches of vacuum water gauge in less than 375 seconds and will maintain greater than or equal to 0.25 inches of vacuum water gauge at a flow rate not exceeding 3324 cubic feet per minute.

The acceptance criteria for the functional test required at least 0.50 inches of vacuum water gauge indicated on control room recorder {BH/PR} PDR-9426A and at least 0.48 inches of vacuum water gauge on control room recorder {BH/PR} PDR-9426B. The two criteria differ to account for the difference in elevation of the reactor building penetrations for the associated sensing lines. The acceptance criteria were selected to ensure the minimum pressure difference across all locations of the reactor building wall will be greater than or equal to 0.25 inches of vacuum water gauge under all postulated environmental conditions.

Both reactor building differential pressure controllers {BH/PDIC} were found to be set incorrectly to 0.50 inches of vacuum water gauge. Per the system operating procedure, they should be set to 0.55 inches of vacuum water gauge. The controllers were reset to the correct setpoint and the reactor building integrity functional test was completed satisfactorily.

Based on a review of maintenance records, PSEG Nuclear believes this condition existed for a time longer than permitted by Technical Specifications. As a result, neither Technical Specification Limiting Condition for Operation 3.6.5.1 for SECONDARY CONTAINMENT INTEGRITY nor the associated ACTION statements were satisfied. This event is reportable as a condition prohibited by plant Technical Specifications in accordance with 10 CFR 50.73(a)(2)(i)(B).

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DESCRIPTION OF OCCURRENCE (continued)

An eight hour notification was made to the NRC in accordance with 10 CFR 50.72(b)(3)(v)(C) on October 8, 2001 at 0833. However, PSEG Nuclear has concluded that, under the postulated worst case environmental conditions, the FRVS would still function to maintain the reactor building at a negative pressure with respect to the outdoors. Therefore this event is not reportable as a condition that could have prevented the fulfillment of the safety function of structures or systems that are needed to control the release of radioactive material. PSEG Nuclear hereby retracts the notification made on October 8, 2001.

APPARENT CAUSE OF OCCURRENCE

Based on a review of maintenance records, PSEG Nuclear believes the reactor building differential pressure controllers were set incorrectly due to personnel error during a maintenance or surveillance testing activity performed at some time after the previous reactor building integrity functional test completed on April 21, 2000.

SAFETY SIGNIFICANCE AND IMPLICATIONS

There were no safety consequences associated with this event. With the as-found FRVS controller setpoints, with the worst case postulated temperature difference, FRVS would maintain a negative pressure across all locations of the reactor building wall respect to the outdoors. It is possible that vacuum in the upper portions of the reactor building (above the 174-foot elevation) could be slightly less than 0.25 inches water gauge. However, PSEG Nuclear previously concluded after a design review and field walkdown that there are no credible leakage paths in the reactor building above the 174-foot elevation (the building is comprised of a concrete shield building with a welded steel liner). Therefore, under postulated worst case temperature differences, all primary containment leakage would still exhaust through the filtered FRVS pathway. There was no impact to the public health and safety.

This event does not constitute a Safety System Functional Failure (SSFF) as defined in NEI 99-02.

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

PREVIOUS OCCURRENCES

A review of previously reported events for the last two years identified two similar events involving component misadjustment or incorrect setting due to personnel error. LER 354/00-009-00 reported an event in which an improperly installed locking device allowed a common supply damper in the Filtration, Recirculation, and Ventilation System (FRVS) to drift closed. LER 354/2001-002-00 reported an event in which a relief valve with an incorrect lift setpoint was installed in the Safety Auxiliaries Cooling System. The corrective actions taken for these previous events were focused on non-licensed operators and on control of relief valve maintenance activities and would not have been expected to preclude this event from occurring.

CORRECTIVE ACTIONS

1. Upon discovery, the reactor building differential pressure controller setpoints were restored to the correct values and the reactor building integrity functional test was completed satisfactorily.

COMMITMENTS

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