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DEC 0 7 2001

PSEG
Nuclear LLC

LRN-01-0408

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

Gentlemen:

LER 354/2001-006-00
HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NO. NPF-57
DOCKET NO. 50-354

Gentlemen:

This Licensee Event Report entitled "Discovery of a Pressure Boundary leak During the Outage" is being submitted pursuant to the requirements of 10CFR50.73(a)(2)(ii)(A). The attached LER contains no commitments.

Sincerely,

D. F. Garchow
Vice President - Operations

Attachment

/EHV

C Distribution
LER File 3.7

IE22

NRC FORM 366 (7-2001)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

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

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.

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
HOPE CREEK GENERATING STATION	05000354	1 OF 4

4. TITLE
Discovery of a Pressure Boundary leak During the Outage

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
10	10	01	2001	– 006	– 00	12	07	01	FACILITY NAME	DOCKET NUMBER

9. OPERATING MODE	10. POWER LEVEL
3	

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply)

	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)		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)	x	50.73(a)(2)(ii)(A)		50.73(a)(2)(viii)(B)		

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER (Include Area Code)
E. H. Villar, Licensing Engineer	856-339-5456

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				

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

At approximately 0332, on October 10, 2001, while performing a primary containment walk down at the beginning of the 10th refueling outage, plant personnel observed a leak on the 'A' Reactor Recirculation {AD} Pump {PMP} suction pipe elbow taps. The observed leak was producing a 3-4 inch spray with the reactor vessel pressure at approximately 300-400 psig. The leak was located where a one-inch pipe is welded to the 28-inch suction line of the "A" recirculation pump. Further investigation revealed that the leak was coming from the weld area and was, therefore, a through-wall leak breach of the reactor coolant system pressure boundary. The apparent cause of the leak was attributed to a weld failure due to vibration induced fatigue of the weld. Some of the corrective actions taken were: (1) Station personnel walked down (a) all recirculation lines for any other failure indications and (b) the equipment in the area of the leak was inspected to ensure no damage from leak impingement, (2) Performed technical evaluation of leak – potential for full failure with "A" recirculation pump in service, (3) Performed radiographic examination (RT's) on other extrados lines and penetrant tests (PT's) on all other susceptible welds – for extent of condition, and (4) Fix the cracked weld. This event was reported in accordance with 10CFR50.72.(b)(3)(ii).

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)

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

IDENTIFICATION OF OCCURRENCE

Event Date: October 10, 2001
Discovery Date: October 10, 2001

CONDITIONS PRIOR TO OCCURRENCE

The plant was in OPERATIONAL CONDITION 3 (hot shutdown) for Hope Creek's 10th refueling outage.

No structures, systems, or components were inoperable at the time of the occurrence that contributed to the event.

DESCRIPTION OF OCCURRENCE

At approximately 0332, on October 10, 2001, while performing a primary containment walk down at the beginning of the 10th refueling outage, plant personnel observed a leak on the 'A' Reactor Recirculation {AD} Pump {PMP} suction pipe elbow taps. The observed leak was producing a 3-4' spray with the reactor vessel pressure at approximately 300-400 psig. The leak was located where a one-inch pipe is welded to the 28-inch suction line of the "A" recirculation pump. Further investigation revealed that the leak was coming from the weld area and was, therefore, a through-wall leak of the reactor coolant system pressure boundary.

Upon discovery of the nature of the leak, the control room operating crew entered Technical Specification (T.S.) 3.4.3.2 "Reactor Coolant System – Operational Leakage." T.S. 3.4.3.2 requires that with any pressure boundary leakage, the unit be placed in HOT SHUTDOWN within 12 hours and IN COLD SHUTDOWN within the next 24 hours. At the time the leakage was identified, Hope Creek Station was already in Mode 3 (HOT SHUTDOWN) with the reactor coolant system at a pressure of approximately 300 to 400 psig. The unit achieved COLD SHUTDOWN on October 10, 2001 at approximately 0917 hours well within the Technical Specifications requirement.

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

DESCRIPTION OF OCCURRENCE (cont'd)

This event was reported in accordance with the requirements of 10CFR50.72(b)(3)(ii), and it is being reported in accordance with 10CFR50.73(a)(2)(ii).

APPARENT CAUSE OF OCCURRENCE

The apparent cause of the leak was attributed to weld failure due to vibration-induced fatigue of the weld. The fatigue induced failure was most likely caused by the second natural frequency of the piping with the accelerometer weight being resonant with the five vane passing running frequency of the "A" Recirculation pump.

In 1990, following a few similar weld failures at Hope Creek, PSEG commissioned an independent contractor to review the stress levels in the recirculation piping system. The stress levels were reported to be satisfactory; however, in 1991, the recirculation system was instrumented with accelerometers. As a result of the data collected modifications to the system were performed, although, the data collected did not indicate that the fatigue stress levels were above the endurance limit of the material. The testing was completed, but the accelerometer(s) was not removed. The accelerometers themselves caused a resonance condition to occur in the piping, which led to the identified failure.

SAFETY SIGNIFICANCE AND IMPLICATIONS

There were no actual consequences and no impact to the health and safety of the public or plant personnel. The condition did not result in a pipe break; and there was no radioactive release.
The through wall leak was discovered while the reactor was shutdown during a drywell inspection prior to plant cool down in support of RF10. However, assuming the leak existed prior to plant shutdown and had remained undetected the overall leakage into the drywell would not have exceeded the Technical Specification integrated leakage rate limit. Any leakage of radioactivity into the drywell in either gaseous or liquid form would have been contained by the drywell systems as per design and any subsequent release of this leakage through the plant radwaste systems would have been well within Technical Specification effluent limits.

FACILITY NAME (1)	DOCKET (2) NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
HOPE CREEK STATION	05000354	2001	0 0 6	00	4 OF 4

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

SAFETY SIGNIFICANCE AND IMPLICATIONS (cont'd)

Had leakage from the undetected crack approached the unidentified leakage limit, it would have been detected by the drywell leak detection system, and the actions required by Technical Specifications would have resulted in the detection of the source and its correction.

Had the crack progressed to a complete line failure, the resultant loss of coolant (LOCA) event would have been bounded by the small break LOCA analysis in the Updated Final Safety Analysis Report (UFSAR).

PREVIOUS OCCURRENCES

A review of events over the past two years identified no reportable events due to vibration induced fatigue failure of welds.

CORRECTIVE ACTIONS

1. The accelerometers were removed during the outage.
2. Station personnel walked down all recirculation lines for any other failure indications.
3. The equipment around the area of the leak was inspected to ensure no damage from leak impingement.
4. Performed radiographic examinations (RT's) on other extrados lines and penetrant tests (PT's) on all other susceptible welds – for extent of condition. These examinations were satisfactory.
5. The cracked weld and the affected section of pipe were removed and replaced.
6. The potential to develop an ISI weld inspection plan for RF11 outage will be evaluated by the PSEG Engineering department.
7. This event will be included in the PSEG Operating Experience program for potential improvements in our procedures or processes.

COMMITMENTS

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