CONNECTICUT YANKEE ATOMIC POWER COMPANY

HADDAM NECK PLANT
362 INJUN HOLLOW ROAD • EAST HAMPTON, CT 06424-3099

September 26, 1996
Re: 10CFR50.73(a)(2)(i)
B15921

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

Reference: Facility Operating License No. DPR-61
Docket No. 50-213
Reportable Occurrence LER 50-213/96-019-00

This letter forwards the Licensee Event Report 96-019-00, required to be submitted, pursuant to the requirements of the Haddam Neck Plant's Technical Specifications.

Very truly yours,

J. J. LaPlatney
Unit Director

JJL/eda

Attachment: LER 50-213/96-019-00

cc: Mr. H. J. Miller
Regional Administrator, Region I
475 Allendale Road
King of Prussia, PA 19406

Mr. William J. Raymond
Sr. Resident Inspector
Haddam Neck

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1028-3 REV 2-91

NRC FORM 366 (5-92)

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 5/31/95

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Haddam Neck	05000 -213	1 OF 4

TITLE (4): Pinhole Leak on Inlet Valve to 'A' RHR Heat Exchanger

EVENT DATE (5)			LER NUMBER (6)			REPORT NUMBER (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
08	31	96	96	-- 019 --	00	09	26	96		05000
										05000

OPERATING MODE (9): 5

POWER LEVEL (10): 000

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

	20.402(b)		20.405(c)		50.73(a)(2)(iv)		73.71(b)
	20.405(a)(1)(i)		50.36(c)(1)		50.73(a)(2)(v)		73.71(c)
	20.405(a)(1)(ii)		50.36(c)(2)		50.73(a)(2)(vii)		OTHER
	20.405(a)(1)(iii)		50.73(a)(2)(i)		50.73(a)(2)(viii)(A)		(Specify in Abstract below and in Text, NRC Form 366A)
	20.405(a)(1)(iv)	X	50.73(a)(2)(ii)		50.73(a)(2)(viii)(B)		
	20.405(a)(1)(v)		50.73(a)(2)(iii)		50.73(a)(2)(x)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Stephen Willard, Associate Engineer	(860) 267-2556

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
B	BP	ISV	A200	Y					

SUPPLEMENTAL REPORT EXPECTED (14)

				EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
X	YES (If yes, complete EXPECTED SUBMISSION DATE)		NO		11	15	96

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

On August 31, 1996, at 1050 hours, with the plant in Mode 5 (cold shutdown), a plant operator, during a routine inspection, identified a pinhole leak in the body of an eight inch inlet isolation valve (RH-V-791A) to the 'A' residual heat removal (RHR) heat exchanger. A small buildup of boric acid on the valve body was noted and when it was wiped off a small amount of water (<0.1 ml/min) weeped from the valve. In accordance with ASME Code section XI guidance the valve was declared inoperable. The plant has been shut down since July 22, 1996 (LER 96-013-00) and is currently in a refueling and maintenance outage. The RHR system was inservice at the time of the event. Initial corrective action consisted of placing the 'B' RHR heat exchanger in service and isolating the 'A' RHR heat exchanger. RH-V-791A was closed and since the leak was in the neck area of the valve, above the disc, the weeping stopped. A radiographic examination of the valve was performed and no significant structural defects were identified. A root cause evaluation is in progress and the results of the investigation and associated corrective action will be submitted in a supplemental report. The reactor core will not be offloaded until a relief request from Section XI for RH-V-791A is granted. Once the reactor is defueled the valve will be repaired or replaced.

REQUIRED NUMBER OF DIGITS/CHARACTERS FOR EACH BLOCK

BLOCK NUMBER	NUMBER OF DIGITS/CHARACTERS	TITLE
1	UP TO 46	FACILITY NAME
2	8 TOTAL 3 IN ADDITION TO 05000	DOCKET NUMBER
3	VARIES	PAGE NUMBER
4	UP TO 76	TITLE
5	6 TOTAL 2 PER BLOCK	EVENT DATE
6	7 TOTAL 2 FOR YEAR 3 FOR SEQUENTIAL NUMBER 2 FOR REVISION NUMBER	LER NUMBER
7	6 TOTAL 2 PER BLOCK	REPORT DATE
8	UP TO 18 -- FACILITY NAME 8 TOTAL -- DOCKET NUMBER 3 IN ADDITION TO 05000	OTHER FACILITIES INVOLVED
9	1	OPERATING MODE
10	3	POWER LEVEL
11	1 CHECK BOX THAT APPLIES	REQUIREMENTS OF 10 CFR
12	UP TO 50 FOR NAME 14 FOR TELEPHONE	LICENSEE CONTACT
13	CAUSE VARIES 2 FOR SYSTEM 4 FOR COMPONENT 4 FOR MANUFACTURER NPRDS VARIES	EACH COMPONENT FAILURE
14	1 CHECK BOX THAT APPLIES	SUPPLEMENTAL REPORT EXPECTED
15	6 TOTAL 2 PER BLOCK	EXPECTED SUBMISSION DATE

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10	3	POWER LEVEL
11	1 CHECK BOX THAT APPLIES	REQUIREMENTS OF 10 CFR
12	UP TO 50 FOR NAME 14 FOR TELEPHONE	LICENSEE CONTACT
13	CAUSE VARIES 2 FOR SYSTEM 4 FOR COMPONENT 4 FOR MANUFACTURER NPRDS VARIES	EACH COMPONENT FAILURE
14	1 CHECK BOX THAT APPLIES	SUPPLEMENTAL REPORT EXPECTED
15	6 TOTAL 2 PER BLOCK	EXPECTED SUBMISSION DATE

BACKGROUND INFORMATION

During COLD SHUTDOWN and refueling operations the residual heat removal (RHR) system (EIIS Code:BP) maintains the reactor coolant system (RCS) (EIIS Code: AB) at the appropriate temperature and also ensures that the boron concentration in the RCS will remain equalized. In this mode of operation one of two parallel RHR pumps take a common suction from the RCS loop 1 hot leg and discharge reactor coolant, via a common discharge line, to one of two full capacity tube and shell heat exchangers. These heat exchangers have 8 inch gate valves (EIIS Code: ISV) to isolate both inlet and outlet flows. The coolant is then returned to the RCS loop 2 cold leg.

The RHR system is also used for post accident sump recirculation. When sufficient water is injected from the RWST to fill the containment sump, short term recirculation is entered. In this alignment, one residual heat removal (RHR) pump takes suction from the sump and supplies water to the suction of one HPSI pump, which delivers water to two of the four cold legs. After a predetermined time has elapsed, two path recirculation is initiated. In this alignment, one RHR pump takes suction from the sump and supplies water to the suction of one charging pump, which delivers water at high pressure to the loop two cold leg. The RHR pump also supplies low pressure water directly to the upper reactor vessel head (core deluge).

EVENT DESCRIPTION

On August 31, 1996, at 1050 hours, with the plant in Mode 5 (cold shutdown), a plant operator, during a routine inspection, identified a pinhole leak in the body of an eight inch inlet isolation valve (RH-V-791A) to the 'A' residual heat removal (RHR) heat exchanger. A small buildup of boric acid on the valve body was noted and when it was wiped off a small amount of water (<0.1 ml/min) weeped from the valve. In accordance with ASME Code section XI guidance the valve was declared inoperable. The plant has been shut down since July 22, 1996 (LER 96-013-00) and is currently in a refueling and maintenance outage. The RHR system was inservice at the time of the event. Initial corrective action consisted of placing the 'B' RHR heat exchanger in service and isolating the 'A' RHR heat exchanger. RH-V-791A was closed and since the leak was in the neck area of the valve, above the disc, the weeping stopped.

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

CAUSE OF THE EVENT

A radiog aphic examination of the valve was performed and no significant structural defects were identified.

A root cause evaluation is in progress and the results of the investigation and associated corrective action will be submitted in a supplemental report.

SAFETY ASSESSMENT

This event was reported under 10CFR50.73(a)(2)(ii) as any event, found while the reactor was shut down, that, had it been found while the reactor was in operation, would have resulted in the plant, including its principal safety barriers, being seriously degraded.

Assuming this was a previous existing condition, this event is also reportable under 10CFR50.73(a)(2)(i)(B) as a condition prohibited by the plant's Technical Specifications. Technical Specification 3.5.1.a requires both trains of RHR operable in Modes 1,2 and 3.

Assuming this condition existed while the plant was in Mode 1,2 or 3 and a design basis loss of coolant accident (LOCA) occurred, a leakage path outside of containment would have existed. However, the performance of the RHR system would not have been challenged due to the small magnitude of the leakage (<0.1 ml/min). In addition, the radiological consequence analysis for a large break LOCA assumes a total system leakage of 3 liters/hour which is much higher than the actual amount of the valve leakage.

Based on the above, the safety significance of this event is judged to be low.

CORRECTIVE ACTION

A relief request from ASME Code Section XI is being filed to declare the valve, which is unisolable, operable but degraded. This will allow the reactor to be defueled. Once the reactor is defueled the valve will be repaired or replaced.

NRC FORM 366A (5-92) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104 EXPIRES 5/31/95

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Haddam Neck	05000-213	96	- 019 -	00	4 OF 4

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

ADDITIONAL INFORMATION

Component	Manufacturer	Model No.
8" stainless steel gate valve	Aloyco (Crane)	2216-SP

Commitments

The following are commitments made within this report. All other statements are for information only

B15921-1 A root cause evaluation is in progress and the results of the investigation and associated corrective action will be submitted in a supplemental report.

B15921-2 The reactor core will not be offloaded until the status of RH-V-791A is dispositioned. Once the reactor is defueled the valve will be repaired or replaced.

PREVIOUS SIMILAR EVENTS

None.