CONNECTICUT YANKEE ATOMIC POWER COMPANY

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

October 11, 1996
Re: 10CFR50.73(a)(2)(v)
B15936

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 50-213/96-021-00

This letter forwards the Licensee Event Report 96-021-00, 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 50-213/96-021-00

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

IE22

1028-3 REV 2-91

NRC FORM 366 (5-95) 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)

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

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Haddam Neck	05000 -213	1 OF 7

TITLE (4)
Valve Leakage Results in Nitrogen Intrusion Into RCS During Cold Shutdown

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
										05000
08	28	96	96	-- 021 --	00	10	11	96	FACILITY NAME	DOCKET NUMBER 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)	X	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)		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)
Michael Marino, Technical Support Engineering	(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

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

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

On August 28, 1996, at approximately 0815 hours, with the plant in Mode 5 (cold shutdown), an operator inappropriately repositioned a valve, while verifying a boration flow path, which allowed nitrogen to be injected into the charging system and ultimately collect in the reactor vessel head. The nitrogen addition was immediately detected by increasing pressurizer level and the valve was closed but continued to leak, undetected, by its seat, resulting in continued nitrogen flow to the vessel head. On September 1, 1996, while investigating the excessive use of nitrogen, operators isolated nitrogen to the volume control tank (VCT) causing VCT pressure to drop. This caused pressurizer level to decrease. Upon investigation it was determined that a large nitrogen bubble had accumulated in the reactor vessel head. Also, the vessel level instrumentation was affected by the nitrogen bubble resulting in inaccurate level indication. The root causes of the event were lack of performance of an effective pre-job briefing and an ineffective assessment of the impact of repositioning the valve. Corrective actions will be based, in part, on the results of the root cause evaluation. On September 11, 1996, with the plant in Mode 5, this event was determined to be reportable.

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

BACKGROUND INFORMATION

During Mode 5 depressurized operations, indication of actual level in the reactor vessel is provided by the reactor vessel level indication system (RVLIS) which uses two probes, each with a series of thermocouple pairs, one heated and one unheated, to detect level in increments. As part of the reactor disassembly refueling sequence RVLIS is disconnected to facilitate missile shield removal.

An additional indication of vessel level is the cold calibrated pressurizer level instrument. This is an accurate measurement of vessel level only when the vessel is adequately vented.

A third method is the cavity level indication system (CLIS). This system uses a pressure transmitter attached to a reactor vessel nozzle which is compensated electronically by a vacuum transducer, designed to sense head vacuum.

Core exit thermocouples (CET) provide indication of the temperature at the top of the core.

In Mode 5 several points in the reactor coolant system (RCS) (EIIS Code: AB) are vented through the vent header system. This provides an approximately 10" water vacuum to allow the removal of any noncondensible gases which come out of solution and directs them to a controlled discharge at the plant stack. The stack effluent is monitored by two plant radiation monitors (EIIS Code: IL).

EVENT DESCRIPTION

On August 28, 1996, at approximately 0815 hours, with the plant in Mode 5 (cold shutdown), an operator inappropriately opened valve BA-V-355, while verifying a boration flow path, which allowed nitrogen in the VCT to be injected into the charging system and ultimately collect in the reactor vessel head. The volume was relatively small and the problem was believed to have been corrected by closing the valve and restoring the previous lineup. The charging header was subsequently vented of gas to ensure boration flow path capability. However, although BA-V-355 was closed it continued to leak by its seat, unknown to the operators, resulting in continued nitrogen flow to the vessel head.

On August 29, 1996 reactor vessel level was lowered in preparation for reactor vessel disassembly by draining an estimated 5000 gallons of inventory from the RCS. The missile shield, along with the RVLIS and CET indication had been previously removed. Subsequently operators noticed pressurizer and cavity level slowly increasing which was incorrectly attributed to loop stop valve leakage whereby the primary side of the steam generators drains back to the reactor vessel. The loop stop valves that were previously closed electrically were seated manually in an effort to stop the suspected in-leakage. On August 30, 1996 plant management decided to delay further vessel work and 1000 gallons of water was added to the vessel which restored level in the pressurizer to the approximate value prior to the drain down. No other additions to or drainage from the RCS occurred until September 1, 1996. Cavity level indication continued to slowly increase throughout this period, eventually stabilizing early on September 1, 1996.

On September 1, 1996, while investigating the excessive use of nitrogen, operators isolated the nitrogen supply to the VCT which resulted in an immediate decrease in pressurizer level and VCT pressure. The nitrogen supply was quickly reopened in an unsuccessful attempt to stop the apparent pressurizer level decrease. In parallel with entry into abnormal operating procedure AOP 3.2-31A, "RCS Leak in Modes 5 and 6", the nitrogen supply to the VCT was again isolated after it was determined a nitrogen gas bubble had accumulated in the reactor head. With the nitrogen isolated, the head vent system slowly bled the accumulated gas from the reactor head. Operators performed controlled make-ups to the RCS as inventory in the pressurizer replaced gas in the reactor head. Approximately 5000 gallons of inventory from the refueling water storage tank were added. Subsequently the charging header was vented and the metering pump was run to verify boration path capability.

CAUSE OF THE EVENT

It was determined that the source of the nitrogen flow was through the closed but leaking BA-V-355. It appears to have started leaking following cycling of the valve on August 28, 1996. Nitrogen usage increased at that point and cavity level indication started drifting unexpectedly. Nitrogen from the VCT flowed into the boric acid system through BA-V-354 (normally open) and to the charging pump suction via BA-V-355. The nitrogen then apparently flowed through the charging

pumps through closed valves CH-FCV-110, 110A (flow control valves that do not totally isolate flow), CH-MOV-292B and C, or their bypass, into the loop 2 cold leg to the reactor vessel head.

A conservative estimate of the nitrogen flow rate into the head (based on nitrogen tank level changes) is 7 scfm. The head and the pressurizer were lined up to the vent header. The vent header is maintained at approximately 10" water vacuum. The head was connected to the vent header via a 50' long 1/2" hose. Calculations show that about a 7' head of water is required to flow 7 scfm of nitrogen through the hose.

Therefore, a bubble formed in the head and increased at a decreasing rate, forcing water from the reactor vessel to the pressurizer until an equilibrium condition existed with pressurizer level (actual and indicated) about 7' above actual vessel level. Equilibrium is indicated by a constant level for several hours prior to the event. The cavity level indication system (CLIS) indicated pressurizer level, not vessel level. This is because the CLIS works by sensing pressure at the loop 1 hot leg and vacuum on the head as indicated by a transducer connected to the downstream side of the vent header hose and electronically summing them. With flow through the hose, the vacuum transducer actually senses vent header and, in this case, pressurizer vacuum, not head pressure. Pressurizer/cavity level indicated a level equivalent to 6" below the top of the head. Actual reactor vessel level at this point was not indicated or known to the operators because of the presence of the bubble. Subsequent calculations show that the vessel level was no lower than 32" below the vessel flange and may have been as high as 12" below the flange.

The root causes of the event were the lack of performance of an effective pre-job briefing leading to the initial cycling of BA-V-355 and an ineffective assessment of the impact of repositioning the valve. Operators failed to recognize the valve was still leaking nitrogen into the RCS after closing it.

Contributing causes were as follows:

- The material condition of BA-V-355 failed to stop flow
- A vent system not designed to handle large quantities of gas
- An improperly designed level indication system
- A lack of operator understanding of the improperly designed level indication system
- Insufficient planning to minimize the time without reactor level indication

In addition, the comprehensive, multi event, root cause analysis identified several
programmatic deficiencies which will be addressed.

SAFETY ASSESSMENT

This event is being reported under 10CFR50.73(a)(2)(v)(B) as any event that alone could have prevented the fulfillment of the safety function of structures or systems that are needed to remove residual heat. The event was terminated by plant operators prior to actually affecting any residual heat removal systems.

There was no direct means of level measurement (no RVLIS) and CET indication had been removed. The cavity level instrumentation was affected by the nitrogen bubble and was providing inaccurate indication. It is possible that absent operator action, if the rate of nitrogen intrusion increased or if the ability of the vent system to vent decreased, that the level in the reactor would decrease to the top of the hot legs. If the nitrogen bubble reached the hot legs then nitrogen could have been drawn into the 'A' RHR pump which could have led to vapor binding and eventual failure of the pump. Although not directly related to this event , the 'B' RHR pump failed to start on September 1, 1996 and a separate root cause evaluation determined that the pump had been inoperable since it was last shut down on August 19,1996. This event is being reported in a separate LER.

The operators responded properly to their indications during the September 1 cavity level indication decrease. However, analysis of the

event shows that actual vessel level was well above the hot leg piping (>3 feet) before the September 1, 1996 event and was increasing during the event. Level would have increased without any operator action. Therefore, operability of the RHR system was never threatened.

Additional calculations indicate that pressurizer level would have had to increase by an additional 27% (from water sluiced from the vessel) before vessel level would have reached the top of the hot leg. This would have required an additional failure (e.g. additional 40% restriction to vent header flow or a 40% increase in the nitrogen flow rate) as well as gross operator inattention to threaten RHR operability.

In the event of a loss of the RHR pumps operators would enter Abnormal Operating Procedure AOP 3.2-12, "Loss of Residual Heat Removal System". This procedure provides guidance on aligning an alternate cooling method or on aligning a makeup water source to the reactor vessel when an alternate cooling method is not available to increase the time to boiling in the core. Inventory could have been added using the low pressure safety injection system. At the time of the event all four reactor coolant loops were available for natural circulation upon opening the loop stop valves. Heat removal would have occurred through use of the steam generators via natural circulation or reflux cooling even if nitrogen entered the steam generator tubes. Boiling would have occurred in the core but no fuel damage would result.

Additionally, as a matter of routine Mode 5 and Mode 6 operations, operators were tracking open RCS and containment penetrations with individuals assigned to ensure timely closure well in advance of the RCS exceeding 200 degrees F or reaching saturation conditions.

The safety significance of this event, as it occurred, was therefore low. However, the potential safety significance is high. The fact that operators did not know vessel level is significant. Had a drain down to refueling level been conducted, it is possible that level could have been lowered to the hot leg and RHR would have been lost.

CORRECTIVE ACTION

Training has been provided to operators on this event and on the features and limitations of the cavity level indication system.

The Unit Director issued a written reinforcement of his expectations to all workers of how physical work activities are to be conducted including pre-job briefings.

A jumper to provide RVLIS indication has been installed and the refueling sequence has been changed to minimize the time without RVLIS.

The vent system has been reconfigured with a larger hose, the vacuum compensator has been located closer to the head, and control room indication of the vacuum compensator reading has been provided. This has reduced the potential level error significantly. With the same introduction rate, which gave a seven foot level error, there will now be a less than one foot error with clear indication of a problem.

A list of potential flow paths which may affect RCS inventory or boron concentration in Mode 5 and 6 was developed. Valves in these flow paths are being reviewed to determine their potential impact and material condition. Several valves (e.g. BA-V-355) will be reworked or replaced to ensure their leak tight integrity.

In addition, any programmatic deficiencies identified by the root cause analysis will be resolved.

ADDITIONAL INFORMATION

Commitments

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

B15936-1 Several valves (e.g. BA-V-355) will be reworked or replaced to ensure their leak tight integrity.

B15936-2 Any programmatic deficiencies identified by the root cause analysis will be resolved.

PREVIOUS SIMILAR EVENTS

None.