CONNECTICUT YANKEE ATOMIC POWER COMPANY

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

June 10, 1997
Re: 10CFR50.73(a)(2)(ii)
CY-97-068

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 50-213/96-012-01

This letter forwards the Licensee Event Report 96-012-01, required to be submitted, pursuant to the requirements of the Haddam Neck Plant's Technical Specifications.

Very truly yours,

R.A. Mellor

R.A. Mellor
Director - Site Operations and Decommissioning

RAM/reb

Attachment: LER 50-213/96-012-01

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

NRC FORM 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 04/30/98

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS MANDATORY INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F33), 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	05000213	1 of 4

TITLE (4)
Feedwater Regulating Valve May Not Close During Steam Line Break

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
										05000
06	11	96	96	012	01	06	10	97	FACILITY NAME	DOCKET NUMBER 0500

OPERATING MODE (9)	1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)			
POWER LEVEL (10)	100	20.2201(b)	20.2203(a)(2)(v)	50.73(a)(2)(i)	50.73(a)(2)(viii)
		20.2203(a)(1)	20.2203(a)(3)(i)	X 50.73(a)(2)(ii)	50.73(a)(2)(x)
		20.2203(a)(2)(i)	20.2203(a)(3)(ii)	50.73(a)(2)(iii)	73.71
		20.2203(a)(2)(ii)	20.2203(a)(4)	50.73(a)(2)(iv)	OTHER
		20.2203(a)(2)(iii)	50.36(c)(1)	50.73(a)(2)(v)	Specify in Abstract below or in NRC Form 366A
		20.2203(a)(2)(iv)	50.36(c)(2)	50.73(a)(2)(vii)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Stephen Willard, 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	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE).	X NO				

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

On June 11, 1996, at 2010 hours, with the plant in Mode 1 at 100% power, an engineering analysis determined that the main feedwater regulating valves (FRV) would not fully isolate feedwater flow as required for a main steam line break accident in containment. It was determined that the differential pressure across the valve would overcome the valve spring's closing force. In a design basis steam line break analysis the feedwater motor operated valves (MOV) are required to isolate, however in the event of a single failure of the MOV the FRV is credited with isolation. The failure to isolate feedwater for a steam line break in containment could result in exceeding maximum containment design conditions. The cause of this condition was an erroneous assumption in the accident analysis that the FRVs would isolate against a high differential pressure. Initial corrective action was to station a dedicated licensed operator at the main control board who's sole function is to isolate feedwater after confirming the existence of a faulted steam generator. This supplemental report was issued to document the results of the root cause of the erroneous assumptions in the accident analysis and to retract originally proposed modifications to the feedwater system due to the Haddam Neck plant being in a permanently defueled state.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

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		YEAR		SEQUENTIAL NUMBER		REVISION NUMBER	
Haddam Neck	05000213	96	--	012	--	01	2 of 4

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

BACKGROUND INFORMATION

The main feedwater system (EIIS Code: SJ) supplies water to the four steam generators through four individual main feedwater lines. Each feedwater line has an associated motor operated feedwater isolation valve (MOV) (EIIS Code: ISV), an air-to-open, spring- to- close feedwater regulating valve (FRV) (EIIS Code: FCV) and a manual isolation valve. The feedwater MOVs (FW-MOV-11,12,13,14) receive a signal to close on either a high containment pressure (HCP) signal or a steam generator overfill protection signal. The FRVs (FW-FCV-1301-1,2,3,4) receive a close signal after a 20 second delay on HCP and immediately on an overfill protection signal.

EVENT DESCRIPTION

On June 11, 1996, at 2010 hours, with the plant in Mode 1 at 100% power, an engineering analysis determined that the main feedwater regulating valves (FRV) would not fully isolate feedwater flow as required for a main steam line break accident in containment. It was determined that the differential pressure across the valve would overcome the valve spring's closing force. In a design basis steam line break analysis the feedwater motor operated valves (MOV) are required to isolate, however in the event of a single failure of the MOV the FRV is credited with isolation. The failure to isolate feedwater for a steam line break in containment could result in exceeding maximum containment design conditions.

CAUSE OF THE EVENT

The cause of the event was an erroneous assumption in the accident analysis that the feedwater regulating valves would close and isolate against the differential pressure experienced between an essentially dead-headed main feed pump and a faulted steam generator. This was discovered during the review of a proposed design change to install new electric motors for the main feed pumps.

A root cause analysis on the factors that led to this situation determined that the details of the design change program within the company were vague in requiring positive documentation, and validation and verification of data used in safety related decision making. This allowed unsubstantiated assumptions, which later proved to be inaccurate, to be made in the accident analyses, such as the isolation ability of the feed regulating valves.

SAFETY ASSESSMENT

This event is reportable under 10CFR50.73(a)(2)(ii)(B) as a condition that was outside the design basis accident analysis for the plant. The design basis analysis for a steam line break accident inside containment, coincident with a single failure of the

feedwater motor operated valve, assumes that the feedwater regulating valve would close. An engineering evaluation determined that the regulating valve would be unable to close given the valve characteristics and system parameters. Therefore, feedwater to an affected steam generator would not be isolated and could result in an overpressurization and high temperature event of the containment structure.

The steam line break analysis credits the motor operated isolation valves for termination of feedwater flow to the steam generators. The isolation valves ramp fully closed in 70 seconds. No credit is taken for feedwater flow reduction for 65 seconds upon receiving a safety injection signal from high containment pressure. In fact, single failure of the feedwater regulating valve is assumed in the analysis. This event does not affect the results of the steam line break analysis. The real significance of this event is the loss of single failure protection.

The probability of containment failure as a function of elevated temperatures and pressures have been evaluated for the Probabilistic Risk Assessment (PRA) analysis. Elevated temperature effects on containment for temperatures less than 800 degrees F have little influence on containment response. The median containment pressure for failure is 90 psig which is 225% of the design pressure of 40 psig.

The probability of containment failure remains low given that the following would have to occur coincidentally : initiation of a steam line break event in containment, failure of a motor operated isolation valve to terminate feedwater flow to the affected steam generator, plant operators not responding to continued feedwater flow to the affected generator and a significant increase in containment pressure above the design value.

The ultimate temperature and pressure that would be reached during this postulated event is unknown, however, they are expected to exceed the design basis parameters.

CORRECTIVE ACTION

Immediate corrective action was to station a dedicated, licensed operator at the main control board who's sole function is to verify that the feedwater MOVs go closed following a reactor trip with safety injection/high containment pressure actuation. If this response is not obtained the operator will manually close the MOVs. If isolation cannot be accomplished the operator checks the pressure of the associated steam generator. If steam generator pressure is less than 800 psig the operator is instructed to stop both main feed pumps, place them in trip pull out, and close both main feed pump discharge isolation valves (FW-MOV-15 and 16).

Long term corrective action was to implement modifications prior to or during the refueling outage (RFO 19) which would include automatic shutdown of the main feed pumps and closure of FW-MOV-15 and 16 under specific conditions.

In addition, the feasibility of installing new feedwater regulating valves that would close against the high differential pressure was to be evaluated.

However, since the Haddam Neck Plant has permanently ceased power operation and all fuel has been permanently removed from the reactor vessel,the originally proposed long term modifications will not be implemented.

The Configuration Management Program (CMP) was established at Haddam Neck to correct the detailed design basis to ensure that it fully complies with current plant configuration. Furthermore the Design Control Manual has been thoroughly revised . The changes improve the standards and thoroughness of the design control process and ensure that the new design basis is maintained.

ADDITIONAL INFORMATION

This supplemental report was issued to document the results of the root cause of the erroneous assumptions in the accident analysis and to retract originally proposed modifications to the feedwater system due to the Haddam Neck Plant being in a permanently defueled state.

No commitments were generated as part of this supplemental report.

PREVIOUS SIMILAR EVENTS

None.