NORTHEAST UTILITIES

The Connecticut Light And Power Company
Western Massachusetts Electric Company
Holyoke Water Power Company
Northeast Utilities Service Company
Northeast Nuclear Energy Company

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June 4, 1993
MP-93-447

Re: 10CFR50.73(a)(2)(iv)

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

Reference: Facility Operating License No. DPR-65
Docket No. 50-336
Licensee Event Report 93-004-01

Gentlemen:

This letter forwards Licensee Event Report 93-004-01 submitted as an update to report completion of the corrective actions identified in LER 93-004-00. LER 93-004 was submitted pursuant to 10CFR50.73(a)(2)(iv), reporting any event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature System.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

Stephen Scace

Stephen E. Scace
Vice President – Millstone Station

SES/MB:ljs

Attachment: LER 93-004-01

cc: T. T. Martin, Region I Administrator
P. D. Swetland, Senior Resident Inspector, Millstone Unit Nos. 1, 2 and 3
G. S. Vissing, NRC Project Manager, Millstone Unit No. 2

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NRC Form 366 (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104
EXPIRES: 4/30/92

Estimated burden per response to comply with this information collection request: 50.0 hrs. Forward comments regarding burden estimate to the Records and Reports Management Branch (p-530), U.S. Nuclear Regulatory Commission, Washington, DC 20555, and to the Paperwork Reduction Project (3150-0104), Office of Management and Budget, Washington, DC 20503

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Millstone Nuclear Power Station Unit 2	0 5 0 0 0 3 3 6	1 OF 0 5

TITLE (4): Reactor Trips on Steam Generator Low Water Level

EVENT DATE (5) MONTH	DAY	YEAR	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	REPORT DATE (7) MONTH	DAY	YEAR	OTHER FACILITIES INVOLVED (8) FACILITY NAMES	DOCKET NUMBER(S)
0 2	2 2	9 3	9 3	— 0 0 4 —	0 1	0 6	0 4	9 3		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9)	1
POWER LEVEL (10)	1 0 0

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

20.402(b)	20.402(c)	X 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 (Specify in Abstract below and in Text, NRC Form 366A)
20.405(a)(1)(iii)	50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	
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 AREA CODE	
Robert Borchert, Reactor Engineer, Ext. 4418	2 0 3	4 4 7 - 1 7 9 1

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

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS

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

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

On February 22, 1993, at 0144 hours, with the plant in Mode 1 at 100% power, the "A" steam generator atmospheric steam dump valve (ADV) failed open. During the subsequent transient plant conditions, the "B" main feedwater pump automatically tripped on low suction pressure, and an automatic reactor trip on low steam generator water level occurred at 0151 hours. Operators then performed Emergency Operating Procedure EOP 2525, "Standard Post Trip Actions." All safety related equipment responded as expected and the unit was placed in a stable condition. The cause of the reactor trip was the inability to recover the water level in the "A" steam generator following the trip of the "B" main feedwater pump.

During the subsequent plant startup on February 23, 1993, at 2037 hours, with the plant in Mode 1 at 18% power, high vibration of several main turbine bearings required shutdown of the main turbine. Reactor power was quickly reduced to approximately 14% power and the main turbine was tripped at 2040 hours. During the subsequent transient plant conditions, an automatic reactor trip on low steam generator water level occurred at 2043 hours. Operators then performed Emergency Operating Procedure EOP 2525, "Standard Post Trip Actions." All safety related equipment responded as expected and the unit was placed in a stable condition. The cause of the reactor trip was insufficient feedwater flow to the steam generators for the existing reactor power level.

These events are being reported pursuant to the requirements of Paragraph 50.73(a)(2)(iv), reporting any event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature System.

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

I. Description of Event

On February 22, 1993, at 0144 hours, with the plant in Mode 1 at 100% power, the "A" steam generator atmospheric steam dump valve (ADV) failed open. The ADV failure caused the indicated steam flow rate to decrease and the feedwater level control system to automatically decrease feedwater flow to the "A" steam generator, thus decreasing the steam generator water level. The operators then took manual control of the "A" steam generator feedwater control valve in an effort to restore steam generator water level. During the subsequent transient, the "A" steam generator water level increased to the high level setpoint, causing the feedwater control valve to automatically close and the water level to decrease. After the "A" steam generator feedwater control valve was reopened, excessive feedwater flow rates caused the "B" main feedwater pump to automatically trip on low suction pressure, resulting in an automatic reactor trip on low steam generator water level at 0151 hours. Operators then performed Emergency Operating Procedure EOP 2525, "Standard Post Trip Actions." Auxiliary feedwater automatically initiated with no complications. All safety related equipment responded as expected and the unit was placed in a stable condition.

During the subsequent plant startup, on February 23, 1993, at 2037 hours, with the plant in Mode 1 at 18% power, high vibration of several main turbine bearings required shutdown of the main turbine within 15 minutes. Reactor power was quickly reduced to approximately 14% power and the main turbine was tripped at 2040 hours. Following the main turbine trip, the steam generator pressures increased, causing the differential pressure between the main feedwater pump and the steam generators to decrease, and consequently decreasing the feedwater flow rate to the steam generators. Additionally, the operators were initially concerned about overfeeding the steam generators, and stopped the automatic actions of the feedwater control system and consequently only partially opened the feedwater bypass control valves. When steam generator levels were observed to be decreasing, the feedwater flow rates were then increased. Due to differences in the feedwater bypass valve positions between the two steam generators, more feedwater flow was directed to the "B" steam generator and the "A" steam generator water level did not recover prior to reaching the low level trip setpoint. An automatic reactor trip on low steam generator water level occurred at 2043 hours. Operators then performed Emergency Operating Procedure EOP 2525, "Standard Post Trip Actions." All safety related equipment responded as expected and the unit was placed in a stable condition.

II. Cause of Event

The root cause of the automatic reactor trip on low steam generator water level on February 22, 1993, was the automatic trip of the "B" main feedwater pump on low suction pressure due to the high feedwater flow rates being demanded to recover steam generator water levels.

The cause of the initiating event on February 22, 1993, was the "A" steam generator ADV failing open. The ADV failed open when the spring attachment fitting broke at the point where the spring attaches to the inner diaphragm assembly. The break at the attachment point resulted from a combination of misalignment of the feedback spring and bending of the range spring attachment (possibly during installation), and positioner vibration during normal plant operation. See Figure 1 for a diagram of the ADV valve positioner.

The root cause of the automatic reactor trip on low steam generator water level on February 23, 1993, was insufficient feedwater flow to the steam generators for the existing reactor power level. Several contributing factors were involved in this root cause:

- Following the main turbine trip, steam generator pressures increased, causing the differential pressure between the main feedwater pump and the steam generators to decrease, and consequently decreasing the feedwater flow rate to the steam generators.

- Subsequent to a turbine trip signal, the main feedwater regulating valves automatically close, and the feedwater regulating bypass valves automatically open to approximately 75%. The operators were initially concerned about overfeeding the steam generators, and stopped the automatic actions of the feedwater control system and consequently only partially opened the feedwater regulating bypass valves. When steam generator levels were observed to be decreasing, the feedwater flow rates were then increased. Due to differences in the feedwater bypass valve positions between the two steam generators, more feedwater flow was directed to the "B" steam generator and the "A" steam generator water level did not recover prior to reaching the low level trip setpoint.

A contributing cause to both events was an unfamiliarity by the operators to feedwater transients involving the new steam generators. During licensed operator training on the new steam generators, the operators were informed that the transient water level response of the new steam generators would be more stable when compared to the original steam generators. This statement is correct for water levels greater than 50%, but is incorrect for water levels less than 50%, where the level changes are much more rapid and more indicative of the response observed on the original steam generators.

The cause of the high main turbine bearing vibration appears to be a turbine "rub." The turbine was placed on the turning gear and was restarted on February 24, 1993 with no problems.

III. Analysis of Event

These events are being reported pursuant to the requirements of Paragraph 50.73(a)(2)(iv), reporting any event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature System.

There were no safety consequences from these reactor trip events. All safety related equipment responded as expected and plant operators executed applicable Emergency Operating Procedures accordingly.

IV. Corrective Action

The valve positioner for the "A" steam generator ADV was replaced and the feedback spring was realigned. The valve positioner for the "B" steam generator ADV was replaced, and the as found feedback spring alignment was satisfactory.

Operator shift briefings were conducted to provide information on the observed steam generator level response during the transient conditions. The water level response for the new steam generators, which were installed during the previous refueling outage, is as follows:

- For indicated steam generator water levels greater than 50% (moisture separator region), the indicated level changes are slower when compared to the original steam generators.

- For indicated steam generator water levels less than 50% (downcomer region), the indicated level changes are the same as the original steam generators.

- For indicated steam generator water levels in the downcomer region (less than 50%), the indicated level changes are significantly faster as compared to changes in the moisture separator region (greater than 50%).

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

Additional actions implemented to prevent recurrence were as follows:

a) The ADV Positioners have been added onto the Production Maintenance Management System (PMMS) and will be replaced every outage.

b) All licensed operators have received specific classroom training on the construction of the new steam generators with emphasis placed on how the differences between the old and new steam generators will affect level response.

In addition, all licensed operators have received specific classroom training on Main and Auxiliary Feedwater Controls, followed by simulator training, emphasizing steam generator level response. The simulator model has been upgraded to represent the changes in level control which resulted from the replacement of the steam generators. Normal/routine simulator training will further enhance the operators' response to level transients.

The combination of the classroom and simulator training has provided the operators with the necessary knowledge to accurately control level in the new steam generators.

V. Additional Information

Similar LERS: 87-12, 87-11, 87-09, 87-02

EIIS Codes for referenced components:

- Atmospheric Steam Dump Valve: SB-PCV-C635
- Feedwater Control Valve: SJ-FCV-C635
- Main Feedwater Pump: SJ-P-I075
- Main Turbine: TA-TG-G084

The following component failed during this event:

Atmospheric Steam Dump Valve Positioner

Manufacturer: Moore Industries

Model: 12372-74GS10GC

EIIS Code: SB-0084-M422

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

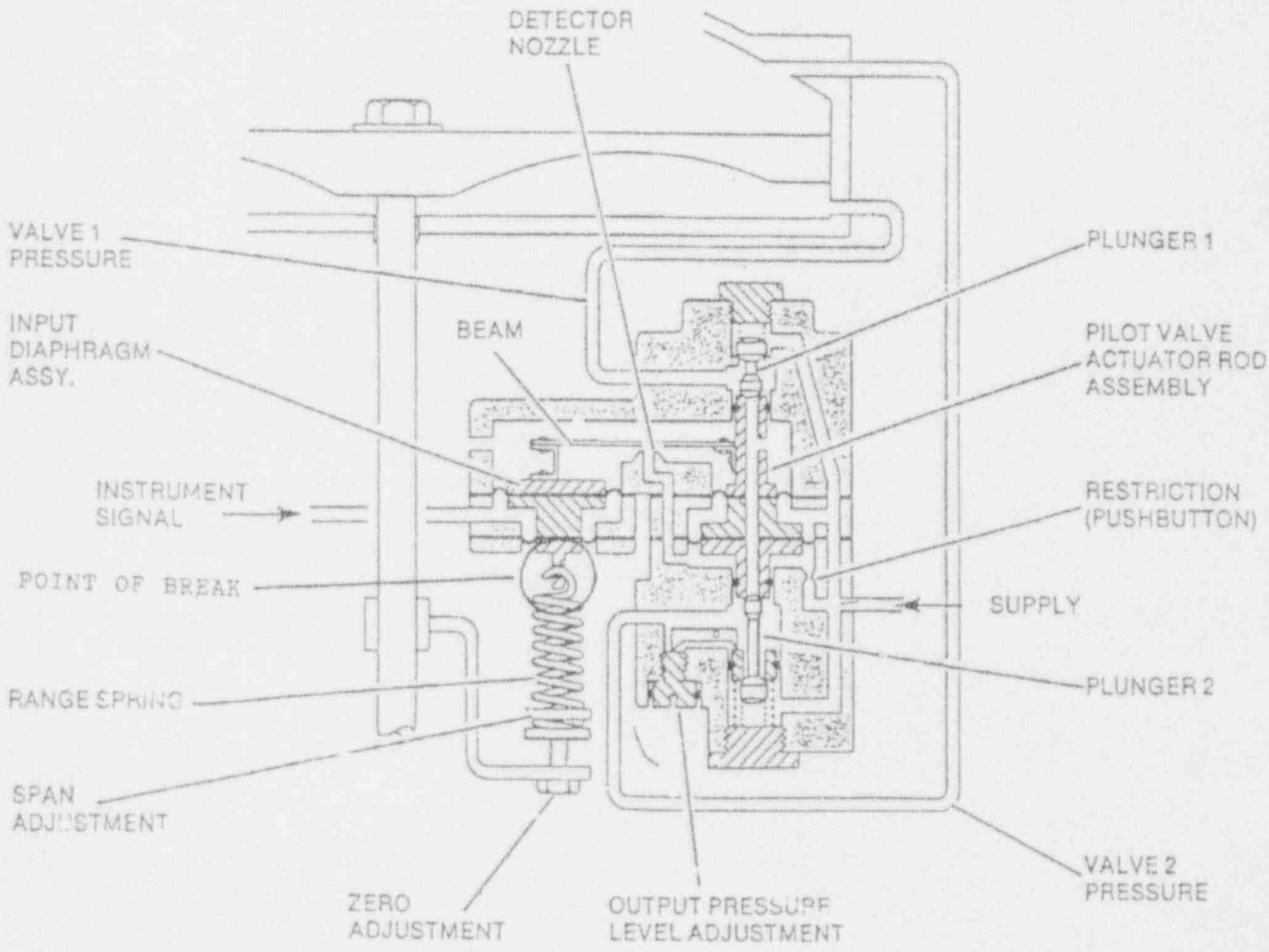

Figure 1
Atmospheric Steam Dump Valve Positioner