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

General Offices · Selden Street, Berlin Connecticut

P.O.BOX 270
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(203)665-5000

October 12, 1990
MP-90-1103

Re: 10CFR50.73

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

Reference: Facility Operating License No. DPR-21
Docket No. 50-245
Licensee Event Report 90-015-00

Gentlemen:

This letter forwards Licensee Event Report 90-015-00 required to be submitted within thirty (30) days pursuant to 10CFR50.73(a)(2)(iv).

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

S. Scace

Stephen E. Scace
Director, Millstone Station

SES/TST:ljs

Attachment: LER 90-015-00

cc: T. T. Martin, Region I Administrator
W. J. Raymond, Senior Resident Inspector, Millstone Unit Nos. 1, 2 and 3
M. Boyle, NRC Project Manager, Millstone Unit No. 1

CertNo P471450906
IE22
1/1

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)	Millstone Nuclear Power Station Unit 1
DOCKET NUMBER (2)	0 5 0 0 0 2 4 5
PAGE (3)	1 OF 0 2
TITLE (4)	Reactor Scram on 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 9	1 4	9 0	9 0	0 1 5	0 0	1 0	1 2	9 0		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9):

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)(iv)	50.73(a)(2)(iii)	50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER AREA CODE	
Trudy Thull, Engineer - Extension 5197	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	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)

YES (If yes, complete EXPECTED SUBMISSION DATE) X NO

EXPECTED SUBMISSION DATE (15) MONTH DAY YEAR

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

On September 14, 1990, with the plant at 100% power (530 degrees Fahrenheit and 1030 psig), a full reactor scram occurred on low reactor water level (+8 inches) after the feedwater regulating valves began to close. The feedwater regulating valves were responding to a high indicated reactor water level signal from the 'A' channel of the feedwater control system, which was controlling feedwater flow. At the time of the scram, a technician was performing a calibration on a pressure switch which senses pressure from an instrument line common to the reference leg of the 'A' feedwater control system.

All safety systems functioned as required and no safety consequences resulted from this event.

NRC Form 366A (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

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

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LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Millstone Nuclear Power Station Unit 1	0 5 0 0 0 2 4 5	9 0	0 1 5	0 0	0 2 OF 0 2

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

I. Description of Event

On September 14, 1990, with the plant at 100% power (530 degrees Fahrenheit and 1030 psig), a full reactor scram occurred on low reactor water level (+8 inches). At the time of the scram, feedwater flow was being controlled by the 'A' channel of the feedwater control system. Indicated level on the 'A' channel of the feedwater control system went high resulting in closure of the feedwater regulating valves. When steam demand exceeded feedwater supply, vessel level decreased to the low reactor water level scram setpoint, resulting in a reactor scram.

II. Cause of Event

During performance of a calibration of pressure switch PS-263-54A, Low Pressure Coolant Injection/Core Spray Reactor Vessel Pressure Switch, the 'A' feedwater control reactor level indication increased. It could not be conclusively proven, based upon followup discussion with the technician and a thorough review of the event, that this calibration caused the event. However, the coincidence of the calibration, together with the absence of any other activities that could have caused the event, led to the conclusion that the indicated level increase was due to a reduction of pressure in a common instrument sensing line. The reduction in pressure of the common instrument sensing line was most likely the result of either a leaking instrument isolation valve or improper valve line up during the calibration. Subsequent investigation has verified that the instrument isolation valve for PS-263-54A had leakage across its seat. However, because the leakage could not be quantified at the exact time of the event, it could not be determined if there was sufficient valve leakage to result in the decrease in pressure of the common sensing line. Discussions with the technician involved indicated that actions taken during the calibration were consistent with guidance provided in departmental instructions for calibration of instruments.

The root cause of the reactor scram has been attributed to the lack of specific procedural guidance for the calibration being performed. Specific guidance should have been provided to detail appropriate precautions and actions, including increased monitoring and awareness, to be taken during the calibration of any instrument located on the common sensing line for the reactor level transmitter.

III. Analysis of Event

All safety systems functioned as required and no safety consequences resulted from this event.

This event is being reported in accordance with 10CFR50.73(a)(2)(iv) which requires the reporting of any event or condition that resulted in a manual or automatic actuation of any Engineered Safeguard Feature (ESF), including the Reactor Protection System (RPS).

IV. Corrective Action

The pressure switch isolation valve will be replaced. Until this valve is replaced, a yellow caution tag has been placed on the valve.

A review has been conducted of all instrument configurations associated with the reactor vessel reference legs, or that have the potential to cause a reactor scram, to ensure that appropriate procedural guidance is provided. Specific to this event, a procedure to calibrate the pressure switch will be developed. All procedures will be implemented by December 31, 1990.

Increased personnel awareness with regards to interaction of instruments and control systems which share common sensing lines has been accomplished by discussions with department personnel and issuance of technical guidance.

V. Additional Information

None