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
HARTFORD, CONNECTICUT 06141-0270
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Re: 10CFR50.73(a)(2)(i)

August 12, 1992
MP-92-875

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 92-013-00

Gentlemen:

This letter forwards Licensee Event Report 92-013-00 required to be submitted within thirty (30) days pursuant to 10CFR50.73(a)(2)(i).

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

FOR: Stephen E. Scace
Vice President - Millstone Station

Harry F. Haynes

BY: Harry F. Haynes
Millstone Unit 1 Director

SES/AAK:ljs

Attachment: LER 92-013-00

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

Cert # P 828 873 581

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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): Millstone Nuclear Power Station Unit 2

DOCKET NUMBER (2): 0 5 0 0 0 3 3 6

PAGE (3): 1 OF 0 3

TITLE (4): Pressurizer Safety Valve Test Failure

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

OPERATING MODE (9): 6

POWER LEVEL (10): 0 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)	[ ] 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)	[X] 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: Albert A. Koehl, Engineer, Ext. 5649

TELEPHONE NUMBER: AREA CODE 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
X	A B	R V	D 2 4 3	Yes					

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 July 13, 1992, Wyle Laboratories performed a lift set point test of Pressurizer Safety Valve (SRV) 2-RC-200 (S/N BN 7128) as required by Technical Specification surveillance 4.4.2. The plant was shut down for refueling with all fuel offloaded to the refuel pool. Both SRVs were removed and sent to Wyle Laboratories for lift set testing. The "as received" test for Valve 2-RC-201 (S/N BR06610) was within the 1% tolerance. The "as received" test for Valve 2-RC-200 (S/N BN7128) recorded set pressures of 2527, 2587, and 2574 psig on three consecutive runs which is above the allowable of 2510 psig. Since the point of discovery was after the unit shut down and no maintenance had been performed on this valve during operation, the period of time in which the unit may have operated outside of Technical Specifications is unknown. An analysis shows that if the Technical Specification set point was exceeded in this case the Primary Coolant System design pressure of 2500 psia +/- 10% would not have been exceeded.

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

I. Description of Event

With the plant shut down for refueling and all fuel removed from the Reactor Pressure Vessel and stored in the fuel pool, both SRVs were removed and sent to Wyle Laboratories for lift set testing as required by Technical Specification Surveillance 4.4.2. On July 13, 1992, Wyle Laboratories performed a lift set point test of SRVs 2-RC-200 (SIN BN 7128) and 2-RC-201 (SIN BRO6610). The "as received" test results for valve 2-RC-201 (SIN BRO6610) were within the 2500 psia (2485 psig) +/- 1% tolerance stated in Technical specification surveillance 4.4.2. The "as received" test results for Valve 2-RC-200.(S/N BN 7128) noted set pressures of 2527, 2587, and 2574 psig on three consecutive runs which exceeded the 2500 psia (2485 psig) +/- 1% tolerance stated in Technical Specification surveillance 4.4.2

II. Cause of Event

The root cause of exceeding the set point tolerance for valve 2-RC-200 is unknown. System vibration and thermal cycling may cause the valves to take a set during long periods of non-use. This is an industry problem which is under investigation.

III. Analysis of Event

The failed valve, S/N BN 7128 was last set and tested on 12/12/89 and installed in the 2-RC-200 location on 10/25/90, during the last refueling outage, therefore it is assumed that the setting drifted out of tolerance during the cycle. Since the point of discovery was after the unit shutdown and no maintenance had been preformed on this valve during operation, the period of time in which the unit may have operated outside of Technical Specifications is unknown. This report is being made in accordance with the requirements of 10CFR50.73(a)(2)(i).

The maximum actuation pressure reported by Wyle Labs. for valve S/N BN 7128 was 2587 psig which is greater than the maximum lift setting allowed by Technical Specification Surveillance Requirements 4.4.2. The maximum setting allowed is 2510 psig (i.e., 2485 psig + 1%). The increase in the valve set point could have raised the peak pressurizer pressure for the limiting design basis pressure transient (i.e. loss of load) above that given in the FSAR. The loss of load analysis predicts a maximum pressure of 2604 psia. The SRVs were assumed to lift at 2575 psia in the analysis. The SRV with the high set point of 2587 psig (2602 psia) would have lifted 27 psi above the FSAR assumed lift setpoint. By adding this increase to the FSAR result, the pressurizer peak pressure due to the drifted SRV is estimated to be 2631 psia. This is conservative because the SRV will relieve a higher flow at higher pressure and hence the over-pressurization would be reduced. Therefore, the limiting peak pressure would be no higher than 27 psi above the design transient peak of 2604 psia or 2631 psia. This pressure is still within the acceptance criterion of 2750 psia (i.e., 110% of the Primary Coolant System design pressure of 2500 psia).

IV. Corrective Action

Following the "As Received" test the valve, S/N BN 7182 is being refurbished, retested and set to 2485 psig +/- 1%.

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

V. Additional Information

Manufacturer	Dresser Industrial Valve and Instrument Company
Type	Spring loaded-balances bellows, enclosed bonnet
Design Pressure	2,485 psig
Design Temperature	675F
Design Code	ASME Section III: Paragraph N 153 in Summer 1969 Addenda; Appendix IX
EIIS Code	XABRVD243Y
Similar Events	LERs 92-010, 90-014, 89-002, 87-014, 86-008, 83-021