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

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April 30, 1993
MP-93-349

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

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

Reference: Facility Operating License No. NPF-49
Docket No. 50-423
Licensee Event Report 93-004-00

Gentlemen:

This letter forwards Licensee Event Report 93-004-00 required to be submitted within thirty (30) days pursuant to 10CFR50.73(a)(2)(iv), any event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS).

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

Stephen Scace

Stephen E. Scace
Vice President – Millstone Station

SES/JSY:ljs

Attachment: LER 93-004-00

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

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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 3	0 5 0 0 0 4 2 3	1 OF 0 4

TITLE (4): Reactor Trip Due to Electro-Hydraulic Control Power Supply 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 5 0 0 0
0 3	3 1	9 3	9 3	— 0 0 4 —	0 0	0 4	3 0	9 3		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)
Jeffrey S. Young, Engineer, Ext. 6442	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	T G	J X	L 0 4 5	Y					
B	S B	R V	D 2 4 3	Y					

SUPPLEMENTAL REPORT EXPECTED (14)

		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
X YES (If yes, complete EXPECTED SUBMISSION DATE)	NO		0 9	0 1	9 3

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

At 0103 on March 31, 1993, with the plant in Mode 1 at 100% power, a turbine valve closure resulted in a reactor trip followed by a turbine trip.

Turbine valve closure was the result of a faulty power supply in the Electro-Hydraulic Control (EHC) system. With the exception of a steam generator code safety valve not completely reseating, the plant responded normally to the transient.

Extensive trouble shooting determined that a power supply in the EHC system was faulty and caused the turbine valves to close. The faulty power supply was replaced. As action to prevent recurrence, the power supplies in the EHC system will be replaced or refurbished on a 10 year period.

Subsequent investigation determined that the steam generator safety valve which did not completely reseat had an incorrect lower adjustment ring setting. Additional inspection revealed that 7 other safety valves also had incorrect settings. Three of these valves indicated that they lifted and reseated during the transient. The other 4 valves did not open.

The root cause of the improper settings has not been determined. Crosby Valve and Gage Company, who performed maintenance on the safeties, documented proper settings and site records do not show any work performed on the safeties after delivery to the site.

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 3	0 5 0 0 0 4 2 3	9 3	— 0 0 4	— 0 0	0 2 OF 0 4

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

I. Description of Event

On March 31, 1993, at 0103 with the plant in Mode 1 at 100% power (586 degrees Fahrenheit and 2250 psia) a turbine load rejection alarm was received as the turbine control valves began to close. A normal response to turbine valve closure occurred. This consisted of all steam dumps opening, atmospheric dumps on 3 steam generators opening, and 1 Pressurizer Power Operated Relief Valve (PORV) cycling open for 6 seconds. In addition, indication was received that several code safety valves on 2 steam generators lifted. Nineteen seconds after the turbine valves began to close, steam generator levels shrank to the low low setpoint which resulted in a reactor trip followed by a turbine trip. After the reactor and turbine trip, the atmospheric dumps on the fourth steam generator opened and indication was received that code safeties on a third steam generator lifted. As steam pressure decreased, all atmospheric steam dumps and safeties reseated with the exception of 1 safety on the "D" steam generator.

At the time of the trip, operators verified that the Reactor Trip and Bypass Breakers were open, that all control rods were fully inserted, and that neutron flux was decreasing. An automatic start of the Auxiliary Feedwater System occurred due to the low low water level in one steam generator and the turbine driven Auxiliary Feedwater Pump started on low low water level in two steam generators. A Feedwater Isolation occurred due to the low average Reactor Coolant System temperature (564 degrees Fahrenheit) coincident with the reactor trip. Subsequently, the plant stabilized at the no load temperature of 557 degrees Fahrenheit. These were expected system responses. No additional Engineered Safety Feature (ESF) actuations were required or initiated.

After the plant had stabilized, the safety was gagged but still did not close completely. A normal plant cooldown was performed and the safety reseated at approximately 1000 psig steam generator pressure.

Subsequent investigation revealed that the lower adjustment ring of the malfunctioning safety valve was incorrect. When set properly, the lower ring insures a clean, forceful popping action when the safety lifts and a cushioned reseating when the safety closes. If this ring is set too high, steam cannot escape as rapidly when the valve begins to close. As a result the blowdown is increased. The settings of the lower adjustment rings on 11 other safeties which had been worked on by Crosby Valve were checked. All 7 of the valves which were installed were found to be set improperly. Of these, 3 indicated that they had lifted and reseated during the transient and 4 did not open because their setpoints were not reached.

II. Cause of Event

The root cause of the turbine valve closure was equipment failure. A capacitor failure in one of the EHC power supplies caused noise to EHC solenoids resulting in a slow closure of all turbine valves with servos.

The root cause of the safety valve failing to completely reseat has not been determined. When the contractor performed maintenance on 4 safeties in 1987 and 1988, NNECO Quality Control performed an inspection of the work on 2 that required ring adjustment. No discrepancies were noted and the ring settings of these safeties were correct. Work on another group of 8 safeties was performed between 1989 and 1991. During this period, the ring settings were documented and a certification statement was issued; however, NNECO inspection was not required as the work was performed under Crosby Valve and Gage Company's Appendix B program. Since plant records indicate that no adjustments were made onsite, a definite cause for the improper adjustment has not been determined. NNECO will continue to work with the contractor to determine when and how the improper adjustments may have occurred.

III. Analysis of Event

This event is being reported in accordance with 10CFR50.73(a)(2)(iv) as any event or condition that resulted in automatic actuation of an ESF including the Reactor Protection System. An immediate notification was made in accordance with 10CFR50.72(b)(2)(ii).

All safety systems functioned as designed as a result of the reactor trip. The Auxiliary Feedwater System started automatically due to the low low steam generator water level. A Feedwater Isolation occurred due to the low Reactor Coolant System average temperature coincident with the reactor trip. No other ESF signals were initiated and the event posed no significant hazard to the health and safety of the public. Secondary plant equipment was returned to normal operation, and the unit was returned to power.

The improper setting of the lower adjustment rings on the safety valves did not place the plant out side of the design basis for a stuck open safety valve. The plant is analyzed for a flow of 977,200 lbm/hr for this accident. Each safety valve is designed to pass 5% (970,000 lbm/hr) of total steam flow. However, parameter indications showed that the affected safety valve had not completely reseated but was not stuck fully open. The partially open safety did not result in an uncontrolled cooldown and the plant operators were able to maintain normal shutdown plant temperature and pressure. Indicated proper operation of 3 other safeties which had improper ring adjustment during the transient showed that improper setting of the lower adjustment ring does not necessarily mean that a safety will not reseat. While actual operation and subsequent testing of the malfunctioning safety (see below) showed that it reseated at lower pressures, these pressures are normally reached several hours after plant shutdown. Therefore, any other valve which did not fully reseat would eventually close by itself.

Test Number	Valve Reseated During Test	Reseat Pressure (PSIG)
1	No	< 1055
2	Yes	1068
3	Yes	1062
4	No	< 1065

Therefore, the 4 valves that did not open may have reseated properly if they had opened. In addition, they would not have stuck fully open if they had failed to reseat and would have closed completely by themselves after a controlled plant cooldown to approximately 530 degrees Fahrenheit.

Because there is no measurable primary to secondary leakage in the "D" steam generator, there was no release of radioactive material as a result of the partially open safety valve.

The implications of the improperly set safety valves on other postulated accidents have not been determined. A review of these implications will be completed and reported in a supplement to this report.

IV. Corrective Action

As immediate corrective action for the trip, the faulty power supply was replaced. As action to prevent recurrence, the power supply will be replaced or refurbished on a 10 year period. Due to replacement unavailability, the other power supplies in the EHC system will be replaced during the upcoming refueling outage.

As immediate corrective action for the improper ring settings, the 11 other safety valves which had been refurbished, repaired or set by Crosby Valve were examined for proper ring settings. The 7 (excluding the malfunctioning one) which were improperly set were adjusted as necessary to their proper ring settings. All spare safeties currently out for maintenance at any contractor will be verified before return to the site.

NRC Form 366A (6-89) U.S. NUCLEAR REGULATORY COMMISSION

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

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

V. Additional Information

No other Licensee Event Reports (LERs) have been submitted for reactor trips resulting from a failed power supply in the EHC system.

This LER also satisfies NNECO's evaluation, notification and reporting obligation to report defects under 10CFR21. In addition, a copy of this LER will be sent to Crosby Valve and Gage Company.

No similar power supply malfunctions were found in a review of NPRDS.

EIIS codes

Systems	Component
Main Turbine Control Fluid System - TG	Power Supply, Electric - JX
Main/Reheat Steam System - SB	Relief Valve - RV