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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Re: 10CFR50.73(a)(2)(iv)
May 7, 1992
MP-92-474

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 92-012-00

Gentlemen:

This letter forwards Licensee Event Report 92-012-00 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).

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

FOR: Stephen E. Scace
Director, Millstone Station

BY: Carl H. Clement
Millstone Unit 3 Director

SES/RNK:ljs

Attachment: LER 92-012-00

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

Cert No
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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 3

TITLE (4): Inadvertent Feedwater Isolation Signal During Reactor Trip Breaker Due to Procedural Inadequacy

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 4	0 7	9 2	9 2	— 0 1 2	— 0 0	0 5	0 7	9 2		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9):

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)	[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 N. Keller, Engineer, Ext. 5507	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 April 7, 1992, at 1601 hours, with the plant in Mode 3 (hot standby), at 0% power, 560 degrees Fahrenheit, and 2260 psia, an inadvertent Feedwater Isolation (FWI) signal was generated due to reactor trip breaker testing. Since the plant was shutdown at the time of the event, and all of the affected isolation valves were closed prior to the event, there was no affect on plant operation.

The root cause of the event is procedural inadequacy. One section of the maintenance procedure for the reactor trip breakers allowed performance of procedural steps out of sequence provided that the procedure intent and technical content would not be changed. This option was exercised and the reactor trip breakers were subsequently put into an alignment which was not normal for performance of the procedure. The FWI signal was generated when a switch inside a reactor trip breaker cubicle was tested. If the procedure sequence was followed as written, these signals would have been blocked.

To prevent recurrence, the maintenance procedure was changed to require performing steps in sequence.

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Millstone Nuclear Power Station Unit 3	0 5 0 0 0 4 2 3	9 2	— 0 1 2 —	0 0	0 2 OF 0 3

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

I. Description of Event

On April 7, 1992, at 1601 hours, with the plant in Mode 3 (hot stand-by), at 0% power, 560 degrees Fahrenheit and 2260 psia, an inadvertent Feedwater Isolation signal was generated. Operators immediately verified that all valves which actuate on an FWI signal were closed. Since the plant was shutdown at the time of the event, and all isolation valves were closed before the FWI signal was generated, there was no affect on plant operation.

At the time the event occurred, a Maintenance technician was performing periodic maintenance (PM) testing on the reactor trip breakers. The control room operators contacted the technician and informed him the FWI signal was generated. It was then determined that the FWI signal was coincident with the technician's testing of a switch mounted inside of the reactor trip breaker cubicle.

II. Cause of Event

The root cause of the event is procedural inadequacy. One section of the maintenance procedure for the reactor trip breakers allowed performance of procedural steps out of sequence provided that the procedure intent and technical content would not be changed. This option was exercised and the reactor trip breakers were subsequently put into an alignment which was not normal for performance of the procedure. The FWI signal was generated when a switch inside a reactor trip breaker cubicle was tested. If the procedure sequence was followed as written, these signals would have been blocked.

The Millstone Unit 3 design is such that an FWI signal is generated after a reactor trip (as sensed by reactor trip/bypass breakers open) if average reactor coolant temperature (Tavg) is 564 degrees Fahrenheit or lower. The inadvertent FWI signal was generated when the technician actuated the "cell" switch, mounted in the rear of the "A" train reactor trip breaker compartment. The reactor trip logic circuit interpreted this switch actuation as having 1 of the 2 reactor trip breakers closed and then re-opened, simulating a reactor trip. Since normal off-line Tavg is below 564 degrees Fahrenheit, an FWI signal was generated. If the procedure sequence was followed as written, the bypass breaker for the "A" train reactor trip breaker would have been closed during the cell switch test, which would have blocked the switch signals.

III. Analysis of Event

This event is being reported under 10CFR50.73(a)(2)(iv), as event or condition that resulted in automatic actuation of an Engineered Safety Feature (ESF). Immediate notifications were performed in accordance with 10CFR50.72(b)(2)(ii).

The reactor trip breaker PMs were being performed with the plant off-line, and with the reactor sub-critical, to avoid risk of potential plant transients. Because of this, all FWI signal actuated components were in their isolation position prior to the event. Additionally, this event could not have occurred with the plant at power due to the required reactor trip breaker alignment. Given these facts, it is concluded that this event posed no significant safety consequences.

IV. Corrective Action

To prevent recurrence of this event, the reactor trip breaker PM procedure has been changed. The note which previously permitted performing procedure steps out of sequence has been removed from the procedure.

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 3	0 5 0 0 0 4 2 3	9 2	— 0 1 2	— 0 0	0 3 OF 0 3

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

V. Additional Information

The following Licensee Event Reports (LERs) document similar incidents in that ESF actuation signals were generated as a result of procedural inadequacies:

LER Number	Subject
87-016	Train 'A' Safety Injection Caused by Instrument Technician Due to Defective Procedure
88-023	Reactor Trip Due to Low Steam Generator Level Due to Main Steam Isolation Valve Closure
88-028	Reactor Trip Due to Loss of Normal 4160V Bus Due to Procedural Deficiency
89-009	Reactor Trip Due to Inadequate Rod Drop Time Recording System Procedure
89-014	Control Building Isolation Actuation Due to Procedural Inadequacy
89-033	Safety Injection on Low Steamline Pressure Due to Procedural Inadequacy

Procedural problems noted in the above listed LERs were in the area of technical guidance in that either no guidance was provided to perform specific evolutions, or the guidance provided in the procedure was inadequate. The event reported in this LER was a result of performing procedural steps out of sequence, as permitted by the procedure. Therefore, the corrective action for the above listed LERs would not have prevented occurrence of the subject event.

EIIS Codes

System Feedwater - SJ

Control Rod Drive - AA

Component

Breaker - BKR

Isolation Valve - ISV