Northeast
Nuclear Energy

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The Northeast Utilities System

Donald B. Miller Jr.,
Senior Vice President - Millstone

Re: 10CFR50.73(a)(2)(4)
10CFR50.73(a)(2)(i)(B)

June 30, 1994
MP-94-433

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 94-009-01

Gentlemen:

This letter forwards update Licensee Event Report 94-009-01.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

Donald B. Miller, Jr.
Senior Vice President - Millstone Station

DBM/RAB:dlr

Attachment: LER 94-009-01

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

IE22
1/1

NRC Form 366 (5-92) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES: 5/31/95

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Millstone Nuclear Power Station Unit 2	05000336	1 OF 4

TITLE (4)
Manual Reactor Trip and Technical Specification Non-Compliance

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 NAME	DOCKET NUMBER
										05000
04	23	94	94	- 009 -	01	06	30	94	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	2	THIS REPORT IS BEING SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)							
			20.402(b)		20.405(c)	X	50.73(a)(2)(iv)		73.71(b)
POWER LEVEL (10)	000		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
			20.405(a)(1)(iii)	X	50.73(a)(2)(i)		50.73(a)(2)(viii)(A)		(Specify in Abstract below and in Text, NRC Form 366A)
			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 (Include Area Code)
Philip J. Lutzi, Site Licensing	(203) 447-1791 Ext. 6585

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	AA	JS	C490	Y						

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 15 single-spaced typewritten lines) (16)

On April 22, 1994, at 2100 hours, with the plant in Mode 1 at 99.8% power, a planned shutdown was commenced to repair a degraded Reactor Coolant Pump seal. At 2113 hours, it was identified that Control Element Assembly (CEA) 7-65 not inserting with the other CEAs in its group. The Shift Supervisor had the indication investigated and declared the reed switch CEA position indicator channel for CEA 7-65 inoperable at 2143 hours and logged into the applicable Technical Specification action statement. The downpower had been terminated when the CEA 7-65 indication problem was identified and was recommenced at 2207 hours.

At 0055 hours on April 23, 1994 with reactor power at approximately 25% and CEA Group 7 at approximately 110 steps, the operators began to suspect that CEA 7-65 was at the fully withdrawn position. At 0115 hours the Shift Supervisor declared CEA 7-65 immovable and logged into the applicable Technical Specification action statement.

At approximately 0230 hours, it was identified that the Technical Specification action statement requirements for an immovable CEA had not been performed within the specified time.

At 0250 hours with the reactor in Mode 2 at approximately 10^{-5}% power and Group 7 CEAs at approximately 90 steps (with the exception of CEA 7-65), power was removed to the Control Element Drive Mechanism (CEDM) for CEA 7-65 and the CEA fully inserted into the core. At 0251 hours the reactor was manually tripped.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Millstone Nuclear Power Station Unit 2	05000336	94	– 009 –	01	02 OF 04

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

I. Description of Event

On April 22, 1994, at 2100 hours, with the plant in Mode 1 at 99.8% power, a planned shutdown was commenced to repair a degraded Reactor Coolant Pump seal. At 2113 hours, while control rods were being inserted for core power distribution control, it was identified that Control Element Assembly (CEA) 7-65 was indicating that it was not inserting.

The downpower was stopped and troubleshooting of CEA 7-65 was performed. This troubleshooting activity indicated that CEA 7-65 was movable and it was believed that the reed switch CEA position indicator channel had become "magnetized" during the long (greater than 150 days) operating run, and that the reed switch channel was inoperable. The Shift Supervisor declared the reed switch CEA position indicator channel for CEA 7-65 inoperable at 2143 hours and logged into the applicable Technical Specification action statement. The downpower was recommenced at 2207 hours using boration.

At approximately 2310 hours, reactor power was less than 70% power as required by the Technical Specification action statement for an inoperable reed switch CEA position indicator channel. The operators began inserting control rods for power distribution control, believing that CEA 7-65 was actually inserting and that the problem was an inoperable reed switch indicator channel.

At 0055 hours on April 23, 1994, with reactor power at approximately 25 % and CEA Group 7 at approximately 110 steps (fully withdrawn position is 176 steps), the operators began to suspect that CEA 7-65 was at the fully withdrawn position. This was based on observed deviations between the four Reactor Protection System (RPS) channels for Nuclear Instrument (NI) power and Axial Shape index (ASI) indications. The Shift Supervisor notified the unit Reactor Engineer and the Duty Officer of the suspected problems with CEA 7-65.

At 0115 hours the Shift Supervisor declared CEA 7-65 immovable and logged into the applicable Technical Specification action statement. At 0150 hours, the Reactor Engineer had confirmed that CEA 7-65 was fully withdrawn based on information from the fixed incore detector monitoring system.

Technical Specification Action Statement (TSAS) 3.1.3.1.a states:

"With one or more full length CEAs inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in at least HOT STANDBY within 6 hours."

Contrary to the above requirements, no actions had been performed to ensure that the shutdown margin requirements were satisfied within one hour of declaring CEA 7-65 immovable. There was no procedural guidance to allow the operators to perform this determination, and the Reactor Engineer had informed the Shift Supervisor at approximately 0230 hours that he had insufficient information available to perform the shutdown margin calculation. Technical Specification 3.0.3 was not entered.

At 0250 hours with the reactor in Mode 2 at approximately 10^{-5}% power and Group 7 CEAs at approximately 90 steps (with the exception of CEA 7-65), power was removed to the Control Element Drive Mechanism (CEDM) for CEA 7-65 and the CEA fully inserted into the core. At 0251 hours the reactor was manually tripped, and operators then performed Emergency Operating Procedure 2525, "Standard Post Trip Actions". All safety related equipment responded as expected.

II. Cause of Event

The root cause of the failure of CEA 7-65 to insert upon demand was determined to be an intermittent failure of one (of three) silicon controlled rectifiers (SCRs) in the upper gripper coil power switch. This failure caused the upper gripper coil to remain energized throughout the insert sequence and therefore did not permit CEA 7-65 to be inserted.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Millstone Nuclear Power Station Unit 2	05000336	94	— 009 —	01	03 OF 04

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

When the Coil Power Programmer (CPP) circuit breaker for CEA 7-65 was opened at 0250 on April 23, 1994, the upper gripper coil de-energized, and resulted in CEA insertion into the core. The SCR subsequently cooled down and the CEDM for CEA 7-65 functioned properly during troubleshooting activities performed on April 23, 1994.

The purpose of the manual reactor trip was to place the reactor in a safe, stable condition and to ensure that shutdown margin requirements were satisfied.

The reason for the operators not recognizing that CEA 7-65 was immovable was their belief that the reed switch CEA position indication channel was inoperable. This belief was based upon the fact that a highly experienced Instrument and Controls Specialist had performed the troubleshooting, and that the reed switches have historically been observed to "magnetize" or "hang up" after a long operating run. A contributing cause to this event was inadequate troubleshooting and lack of a formal troubleshooting plan.

III. Analysis of Event

The manual reactor trip is 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 was no safety consequence from the manual reactor trip event. All safety related equipment responded as expected and plant operators executed applicable Emergency Operating Procedures accordingly.

The failure to fully comply with action statement requirements is being reported pursuant to the requirements of Paragraph 50.73(a)(2)(i)(B), "Any operation or condition prohibited by the plant's Technical Specifications."

There was no safety consequence as a result of the failure to fully comply with the action statement requirement for an immovable CEA. This conclusion is based on the following:

(a) The reactor shutdown was completed in a timely fashion and the requirements of Technical Specification LCO 3.0.3 (though not logged into by the operators) were being satisfied during the period of time that CEA 7-65 was declared immovable.

(b) CEA 7-65 was ultimately determined to be trippable by removing power from its CEDM.

(c) A post-event review of the shutdown margin was performed. This evaluation confirmed that adequate shutdown margin was available at all times during the plant shutdown on April 22 - 23, 1994. This evaluation assumed that CEA 7-65 was not trippable and that the most reactive CEA remained in the fully withdrawn position.

(d) A post-event review of the core power distribution parameters during the shutdown have confirmed that the Departure from Nucleate Boiling (DNB) and Linear Heat Rate limits were not exceeded during the period of time that CEA 7-65 was misaligned with the other CEAs in its group.

IV. Corrective Action

The upper gripper power switch module has been replaced for CEA 7-65. The power switch has been sent to the vendor for failure analysis.

Procedure AOP 2556, "CEA Malfunctions" has been revised to provide guidance for each of the Technical Specification action statement requirements for CEA position and CEA position indicator channels. Training on the revised procedure is complete.

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 2	05000336	94	— 009 —	01	04 OF 04

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

The Technical Specifications action statement requirements were reviewed to ensure that adequate procedural guidance exists.

Diagnostic evaluations and refresher training in the areas of reactor theory and Technical Specification knowledge and application have been completed.

A formal troubleshooting plan to evaluate CEA operability and CEA position indication system problems has been developed.

V. Additional Information

Similar LERS:

Manual Reactor Trips: 91-012, 91-004, 90-006, 84-012

Action Statement non-compliance: 93-002, 92-009, 90-002, 89-009.
All of these events involved a failure to log into Technical Specification action statements. The corrective actions in each of these events were unique to the event

EIIS Codes for referenced components:
Control Element Assembly: AA-ROD-C490
Reed Switch CEA position indicator: AA-EIS-C490
CEA Motion Inhibit circuit: AA-DCC-C490
Reactor Protection System: JC-C490
Nuclear instrumentation: IG-C490
Incore Detector Monitoring System: IG-C490
Control Element Drive Mechanism: AA-75-C490
Reactor Coolant System: AB-C490
Silicon Controlled Rectifier: AA-SCR-C490
Coil Power Programmer: AA-STC-C490

The following component failed during this event:

CEA 7-65 Upper gripper power switch: AA-JS-C490

Manufacturer: Combustion Engineering

Model No.: N 9018, Rev. 1

Serial No.: E4049