NRC FORM 366
(7-77)

U. S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT

Update Report
Previous Report Date 10-28-83

CONTROL BLOCK: (1) (PLEASE PRINT OR TYPE ALL REQUIRED INFORMATION)

0 1	LICENSEE CODE (2)	C T M N S 2
	LICENSE NUMBER (3)	0 0 - 0 0 0 0 0 - 0 0
	LICENSE TYPE (4)	4 1 1 1 1
	CAT (5)	
CON'T 0 1	REPORT SOURCE (6)	L
	DOCKET NUMBER (7)	0 5 0 0 0 3 3 6
	EVENT DATE (8)	0 6 3 0 8 3
	REPORT DATE (9)	0 4 1 7 8 4

EVENT DESCRIPTION AND PROBABLE CONSEQUENCES (10)

0 2 During schedule inspections of the reactor vessel core barrel and thermal
0 3 shield (TS), damage to the TS was discovered. In addition, non-destructive
0 4 examination of the core support barrel (CSB) revealed a thru - wall crack at
0 5 two of the nine CSB lugs. There were no consequences, since the damaged TS
0 6 remained in position and there was no blockage of any core flow passages.
0 7 Similar LER's: None
0 8

0 9	SYSTEM CODE (11)	CAUSE CODE (12)	CAUSE SUBCODE (13)	COMPONENT CODE (14)	COMP. SUBCODE (15)	VALVE SUBCODE (16)
	R A	E	B	V E S S E L	A	Z

(17) LER RO REPORT NUMBER	EVENT YEAR		SEQUENTIAL REPORT NO.		OCCURRENCE CODE	REPORT TYPE		REVISION NO.
	8 3	—	0 2 4	/	0 1	X	—	2

ACTION TAKEN (18)	FUTURE ACTION (19)	EFFECT ON PLANT (20)	SHUTDOWN METHOD (21)	HOURS (22)	ATTACHMENT SUBMITTED (23)	NPRD-4 FORM SUB. (24)	PRIME COMP. SUPPLIER (25)	COMPONENT MANUFACTURER (26)
X	Z	C	Z	0 0 0 0	Y	Y	N	C 4 9 0

CAUSE DESCRIPTION AND CORRECTIVE ACTIONS (27)

1 0 The cause of the damage to the TS appears to be the result of hydraulically
1 1 induced loading. The deteriorated TS support system resulted in additional
1 2 loading at the CSB lugs causing the cracks. The TS was removed and the CSB
1 3 was repaired.
1 4

1 5	FACILITY STATUS (28)	% POWER (29)	OTHER STATUS (30)	METHOD OF DISCOVERY (31)	DISCOVERY DESCRIPTION (32)
	H	0 0 0	NA	B	Inservice Inspection

1 6	ACTIVITY RELEASED (33)	CONTENT OF RELEASE (34)	AMOUNT OF ACTIVITY (35)	LOCATION OF RELEASE (36)
	Z	Z	NA	NA

PERSONNEL EXPOSURES

1 7	NUMBER (37)	TYPE (38)	DESCRIPTION (39)
	0 0 0	Z	NA

PERSONNEL INJURIES

1 8	NUMBER (40)	DESCRIPTION (41)
	0 0 0	NA

8404240354 840417
PDR ADOCK 05000336
S PDR

LOSS OF OR DAMAGE TO FACILITY (43)

1 9	TYPE (42)	DESCRIPTION
	Z	NA

IE22 ///

PUBLICITY

2 0	ISSUED (44)	DESCRIPTION (45)	NRC USE ONLY
	Y	Routine Public Relations Releases 7/1/83	

NAME OF PREPARER J.G. Resetar PHONE: 203-447-1791

ATTACHMENT TO LER 83-24/1X-2
NORTHEAST NUCLEAR ENERGY COMPANY
MILLSTONE NUCLEAR POWER STATION - UNIT 2
FACILITY OPERATION LICENSE NUMBER DPR-65
DOCKET NO. 50-336

During a scheduled inspection of the reactor vessel core barrel and thermal shield, damage to the thermal shield was discovered. It was determined that the required course of action would be to remove the thermal shield. As described in the detailed Thermal Shield Damage Recovery Program Report, W.G. Counsil letter to J.R. Miller, dated December 12, 1983, the thermal shield, in the area of the support lugs and support pins, was heavily damaged near the 0° and 180° axis, less damaged near the 90° and 270° axis. Damage to the shield caused both the support pin and large pieces of the thermal shield to break loose at 4 of the lug 9 locations.

Upon removal of the thermal shield, non-destructive examinations using visual eddy current and ultrasonic techniques revealed damage to the core barrel at lug locations 4 and 5. The core barrel was found to be free of any flaws at other lug locations. One through wall crack was found at lug 4 and 5. Both lugs were removed by a machining process and NDT procedures were repeated to confirm that no other cracks were present. Surface flaws were removed by machining followed by eddy current inspection of the slot surface. Through wall flaws were arrested by drilling a 1 - 1/8" diameter hole at each end of the crack. The design considerations for the CSB repair involved fatigue and crack stability. The hole was sized to keep the intensified stresses below the endurance limit from the ASME code curves. With stresses below this endurance limit, the normal operating loads are not expected to propagate a flaw from the arrestor hole.

Extensive analyses were performed to verify acceptable CSB operation. The description and results of these analyses are documented in the above mentioned letter.

The cause of the damage to the thermal shield appears to be the result of hydraulically induced loading. This loading deteriorated the thermal shield support system which resulted in additional loading at the CSB lugs causing the through wall cracks.

The effect of the thermal shield removal and the core support barrel repairs on the core thermal and hydraulic conditions is a slightly higher core flow due to the removal of the thermal shield and a slightly lower core flow due to the increased bypass flow through the crack arrestor holes. The net effect on actual operation is that the core thermal hydraulic conditions changed by an insignificant amount.

The impact of removing the thermal shield on the Millstone 2 docketed safety analyses has been assessed and presented to the NRC Staff. It is concluded that operation without a thermal shield is acceptable based on the results of the currently docketed safety analyses.

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
(203) 666-6911

April 17, 1984
MP-5961

Dr. Thomas E. Murley
Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
631 Park Avenue
King of Prussia, Pennsylvania 19406

Reference: Facility Operating License DPR-65
Docket No 50-336
Reportable Occurrence RO-50-336/83-24/01X-2

Dear Dr. Murley:

This letter forwards updated Licensee Event Report 83-24/01X-2 required to be submitted pursuant to the requirements of Millstone Unit 2 Appendix A Technical Specifications, Section 6.9.1.8.i, discovery of conditions not specifically considered in the Safety Analysis Report or Technical Specifications that require remedial action or corrective measures to prevent the existence or development of an unsafe condition. An additional three copies of the report are enclosed.

Yours truly,

NORTHEAST NUCLEAR ENERGY COMPANY

E. J. Mroczka
Station Superintendent
Millstone Nuclear Power Station

EJM/JGR:ejl

Attachment: LER RO-50-366/83-24/01X-2

cc: Director, Office of Inspection and Enforcement Washington, D. C. (30)

Director, Office of Management Information and Program Control, Washington, D.C. (3)

U.S. Nuclear Regulatory Commission, c/o Document Management Branch, Washington, D.C. 20555

IE22