NRC FORM 366 (7-77)

U. S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT

CONTROL BOCK: [- | | | | |] (1) (PLEASE PRINT OR TYPE ALL REQUIRED INFORMATION)

| 0 1 | LICENSEE CODE: P A T M I 1 (2) | LICENSE NUMBER: 0 0 - 0 0 0 0 0 - 0 0 (3) | LICENSE TYPE: 4 1 1 1 1 (4) | CAT (5) |

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| 0 1 | REPORT SOURCE: L (6) | DOCKET NUMBER: 0 5 0 0 0 2 8 9 (7) | EVENT DATE: 0 9 3 0 8 3 (8) | REPORT DATE: 1 0 1 8 8 3 (9) |

EVENT DESCRIPTION AND PROBABLE CONSEQUENCES (10)

0 2 While shutdown, an evaluation of instrument line seismic and thermal loads, as a
0 3 follow-up to Maintenance Non-conformance Report (MNCR), determined that selected
0 4 instrument lines required modifications in order to meet the plant's original de-
0 5 sign basis. Although the USAS B31.1 Code Stress Criteria were not met, the evalua-
0 6 tion results revealed that the stresses resulting from thermal and seismic loads
0 7 are below the ASME III operability stress criteria. Public health and safety
0 8 were unaffected. Reportable per Technical Specification 6.9.2.A.9.

| 0 9 | SYSTEM CODE: I C (11) | CAUSE CODE: B (12) | CAUSE SUBCODE: A (13) | COMPONENT CODE: Z Z Z Z Z Z (14) | COMP SUBCODE: Z (15) | VALVE SUBCODE: Z (16) |

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

ACTION TAKEN	FUTURE ACTION	EFFECT ON PLANT	SHUTDOWN METHOD	HOURS (22)	ATTACHMENT SUBMITTED	NPRD-4 FORM SUB.	PRIME COMP. SUPPLIER	COMPONENT MANUFACTURER
F (18)	F (19)	Z (20)	Z (21)	0 0 0 0	Y (23)	N (24)	N (25)	B 0 1 5 (26)

CAUSE DESCRIPTION AND CORRECTIVE ACTIONS (27)

1 0 The original analysis of the instrument tubing and supports by the NSSS did not
1 1 account for the thermal growth of the main piping and its impact on these
1 2 instrument lines. Installation of supports and thermal expansion loops for
1 3 instruments indicated in attached Table 1 will be installed prior to criticality.
1 4

| 1 5 | FACILITY STATUS: X (28) | % POWER: 0 0 0 (29) | OTHER STATUS (30): NRC Order | METHOD OF DISCOVERY: C (31) | DISCOVERY DESCRIPTION (32): QA Inspection |

| 1 6 | ACTIVITY RELEASED: Z (33) | CONTENT OF RELEASE: Z (34) | AMOUNT OF ACTIVITY (35): N/A | LOCATION OF RELEASE (36): N/A |

PERSONNEL EXPOSURES

| 1 7 | NUMBER: 0 0 0 (37) | TYPE: Z (38) | DESCRIPTION (39): N/A |

PERSONNEL INJURIES

| 1 8 | NUMBER: 0 0 0 (40) | DESCRIPTION (41): N/A |

LOSS OF OR DAMAGE TO FACILITY (43)

| 1 9 | TYPE: Z (42) | DESCRIPTION: N/A |

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PUBLICITY

| 2 0 | ISSUED: N (44) | DESCRIPTION (45): N/A | NRC USE ONLY |

NAME OF PREPARER C. J. Stephenson PHONE (717) 948-8554

LER No. 83-032

IMPORTANT TO SAFETY INSTRUMENT IMPULSE
TUBING INSIDE THE REACTOR BUILDING

I. Current Activities at the Time of the Occurrence

TMI-1 was in a long term cold shutdown condition.

II. Circumstances Leading to the Occurrence

As a follow-up to a Maintenance Non-Conformance Report (MNCR) on instrument tube sloping, both thermal stress and siesmic stress were reevaluated for all Important to Safety instruments inside the Reactor Building.

III. Description of Occurrence

The thermal expansion and seismic condition stresses have been evaluated using acceptance criteria of USAS B31.1 as defined in FSAR paragraphs 5.1.1 and 5.1.2.1.2 for all Important to Safety Instrument impulse tubes. These stresses calculated by a simplified method were found to exceed the USAS B31.1allowables for the instruments listed in Table 1.

IV. Resultant Events

There have been no seismic events during the lifetime of the plant and no loss of instrument function due to thermal cycling.

V. Previous Events of a Similar Nature

None.

VI. Root Cause of Occurrence

The initial analysis of the instrument tubing and supports did not appear to adequately take into account the thermal growth of main piping and its impact on the seismic design.

VII. Immediate Corrective Action

Because of the impending hot cycle testing, an analysis was performed using simplified methods and concluded that although the tubing appeared to exceed the allowables of USAS B31.1 the testing could continue without concern for failure because:

a. Failure would not occur by fatigue. A typical configuration for existing instrument tubing undergoing a process pipe thermal expansion in the vertical direction was analyzed according to the methods described in Section III Paragraph NB-3228.5 of the ASME Code. The number of cycles which the

instrument impulse tubing would take before failure using the above method is 5500 cycles. Assuming a cycle per week from initial startup testing through power operation, the approximate thermal cycles to date is 350; therefore, the remaining cycles would be approximately 5150 cycles. This would allow many more cycles to occur without undue cause for concern.

b. The seismic capability of the instrument lines is sufficient to preclude loss of function. The deadweight plus seismic stresses in the tubing are well within the operability allowable of 2 Sy, which is defined as service level "D" allowable for Equation 9 of the ASME Code Section III for pipe analysis. This allowable guarantees no loss of operability due to line pinch.

c. The plant will not be taken critical and therefore safety system functions for these instruments will not be required to shutdown the plant.

VIII. Long Term Corrective Action

Installation of supports and thermal expansion loops for instruments indicated in Table 1 will be installed before criticality such that B31.1 is satisfied.

IX. Component Failure Data

None.

TMI-1 INSTRUMENTS INVOLVED IN
INSTRUMENT IMPULSE TUBING
MODIFICATION

System	Instrument Numbers
Main Steam S/G Out Press	SP-6APT-1 SP-6APT-2 SP-6BPT-1 SP-6BPT-2 PS-600, 601, 602 603, 604, 605, 606, 607 PT-950 and PT-951
Main Steam Op S/G Level	SP-1ALT-3, SP-1B-LT-2 SP-1A-LT-2 SP-1B-LT-3
Main Steam Op & Startup S/G Level	SP1B-LT-3 SP1B-LT-5 SP1B-LT-2 SP1B-LT-4 SP1A-LT-2 SP1A-LT-4
Main Steam Start-up S/G Level	SP1A-LT-4&5 SP1B-LT-4&5
Main Steam Full Range S/G Level	SP1A-LT-1 SP1B-LT-1
Reactor Coolant PZR Level Xmitter	RC-1-LT3 RC-1-LT2 RC-1-LT1
Reactor Coolant RC Flow	RC14A-DPT-3 RC14A-DPT-4 RC14A-DPT-1 RC14A-DPT-2 RC14B-DPT-1, 2, 3, 4
Reactor Coolant Narrow/Wide Range RC Press	RC-3A-PT2 RC-3A-PT4 RC-3B-PT1 RC-3B-PT3
Reactor Coolant Narrow Range RC Press	RC-3A-PT1 RC-3A-PT3

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October 18, 1983
5211-83-302

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

Dear Sir:

Three Mile Island Nuclear Station, Unit 1 (TMI-1)
Operating License No. DPR-50
Docket No. 50-289
LER 83-032/01T-0

This letter transmits Licensee Event Report (LER) 83-032/01T-0 which deals with instrument lines which require modifications to meet original Design Basis. The public health and safety were not affected. This report is being submitted late as discussed with R. Conte, Senior Resident Inspector.

Sincerely,

H. D. Hukill
Director, TMI-1

HDH:CJS:vjf

Enclosure

cc: R. Conte
J. Van Vliet
Document Management Branch

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