NRC Form 366
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO 3150-0104
EXPIRES 8/31/85

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
THREE MILE ISLAND, UNIT 1	0 5 0 0 0 2 8 9	1 OF 0 3

TITLE (4)
HIGH PRESSURE REACTOR TRIP CAUSED BY INADEQUATE FEEDWATER FLOW

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	2 3	8 6	8 6	— 0 1 0 —	0 0	0 5	2 2	8 6		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): N
POWER LEVEL (10): 0 0 9

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)

20.402(b)	20.405(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)
M. R. KNIGHT, TMI-1 LICENSING ENGINEER	7 1 7 9 4 8 - 8 5 5 4

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

SUPPLEMENTAL REPORT EXPECTED (14)

YES (If yes, complete EXPECTED SUBMISSION DATE) — X NO

EXPECTED SUBMISSION DATE (15): MONTH / DAY / YEAR

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

TMI-1 was at approximately 9% reactor power during power escalation following startup from the OTSG Eddy Current Outage. At 0510 on April 23, 1986, the reactor tripped on high Reactor Coolant System (RCS) pressure resulting from inadequate main feedwater flow while transferring the main feedwater pump turbine (SJ/TRB)* steam supply from the auxiliary boiler to main steam supply. This event was a result of personnel error in that the manual action taken by the operator during the transition was too slow to maintain adequate feedwater to the OTSGs. As a result, RCS pressure increased to the high pressure trip setpoint and the reactor tripped.

All safety systems functioned as designed and the operating crew brought the plant to Hot Shutdown conditions.

To prevent a recurrence of this event, procedural changes are being made to provide additional guidance to the operator when making the transition from auxiliary (SA/-)* to main steam (SB/-)* supply for the main feedwater pump turbines (SJ/TRB)*. Shift supervisors will review this event with each of the operating crews. In addition, procedural requirements are being incorporated for headset communications between the control room and other stations during plant transitions which require such close coordination.

8605290719 860522
PDR ADOCK 05000289
S PDR

IE22
1/1

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
THREE MILE ISLAND, UNIT 1	0 5 0 0 0 2 8 9	8 6	— 0 1 0 —	0 0	0 2 OF 0 3

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

HIGH PRESSURE REACTOR TRIP CAUSED BY INADEQUATE MAIN FEEDWATER PUMP FLOW DURING THE TRANSITION FROM AUXILIARY TO MAIN STEAM SUPPLY

I. Plant Operating Conditions Before the Event

The plant was at approximately 9% reactor power during power escalation following startup from the OTSG Eddy Current Outage. Preparations for rolling and loading the Turbine Generator were in progress. The operating main feedwater pump (SJ/P)* was under hand control.

II. Status of Structures, Components, or Systems that were Inoperable at the Start of the Event and that Contributed to the Event

None

III. Event Description

On April 23, 1986 while escalating power above 8% following the OTSG Eddy Current Outage, the reactor tripped on high Reactor Coolant System (RCS) pressure at 0510 hours. High RCS pressure resulted from inadequate feedwater flow to the OTSGs. Immediately prior to the reactor trip, the main feedwater pump turbine (SJ/TRB)* steam supply was being transferred from auxiliary boiler (SA/-)* steam to main steam (SB/-)* supply. During the transfer, feedwater pump speed decreased sufficiently to cause inadequate feedwater flow to the OTSGs (AB/SG)*. Although the operator increased feedwater pump speed to restore OTSG feed prior to the trip, the action occurred too late to prevent exceeding the 2300 psig high pressure reactor trip setpoint. Maximum RCS pressure was 2305 psig. Post trip response was normal. The operating crew carried out their appropriate actions in accordance with the Reactor Trip procedure and brought the plant to Hot Shutdown conditions.

Followup investigation concluded that the root cause of the trip was slow operator response time (personnel error).

IV. Component Failure Data

Not Applicable.

V. Automatic or Manually Initiated Safety System Response

All safety systems functioned as designed. The Reactor Protection System (RPS), (JC/-)*, initiated a reactor trip upon reaching the 2300 psig RCS pressure trip setpoint as designed.

VI. Assessment of the Safety Consequences and Implications of the Event

There were no safety implications from this event.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
THREE MILE ISLAND, UNIT 1	0 5 0 0 0 2 8 9	8 6	— 0 1 0	— 0 0	0 3 OF 0 3

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

VII. Previous Events of a Similar Nature

None.

VIII. Corrective Actions Planned

Shift Supervisors have reviewed this event with each of the operating crews.

To prevent a recurrence of this event, procedure changes have been initiated to provide additional guidance to the operator in making the transition from Auxiliary to Main Steam Supply for the main feedwater pump turbines.

In addition, procedural changes have been initiated to incorporate requirements for headset communications between the control room and other stations during plant transitions that require such close coordination.

As required by 10CFR50.73.b.2.ii.F, the IEEE Std. 803-1983 Energy Industry Identification System (EIIS) System Identifier (SI) and Component Function Identifier (CFI) appear in parentheses with an asterisk: (SI/CFI).

GPU Nuclear

GPU Nuclear Corporation
Post Office Box 480
Route 441 South
Middletown, Pennsylvania 17057-0191
717 944-7621
TELEX 84-2386
Writer's Direct Dial Number:

May 22, 1986
5211-86-2096

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

Dear Sir:

Three Mile Island Nuclear Station Unit 1 (TMI-1)
Operating License No. DPR-50
Docket No. 50-289
LER 86-010-00

This letter transmits License Event Report (LER) No. 86-010-00 which deals with a high pressure reactor trip caused by inadequate main feedwater pump flow during power escalation in the transition from auxiliary to main steam supply for the main feedwater pump. Public health and safety were unaffected.

This LER is being submitted pursuant to 10 CFR 50.73, using the required NRC forms (attached). NRC Form 366 contains an abstract which provides a brief description of the event. For a complete understanding of the event, refer to the text of the report which appears on Form 366A.

Sincerely,

H D Hukill

H. D. Hukill
Director, TMI-1

HDH/MRK/spb

Attachment

cc: T. Murley
R. Conte
J. Thoma

0592A

IE22
1/1