

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D.C. 20555

APRIL 5, 1979

IE Bulletin 79-05A

NUCLEAR INCIDENT AT THREE MILE ISLAND - SUPPLEMENT

Description of Circumstances:

Preliminary information received by the NRC since issuance of IE Bulletin 79-05 on April 1, 1979 has identified six potential human, design and mechanical failures which resulted in the core damage and radiation releases at the Three Mile Island Unit 2 nuclear plant. The information and actions in this supplement clarify and extend the original Bulletin and transmit a preliminary chronology of the TMI accident through the first 16 hours (Enclosure 1).

1. At the time of the initiating event, loss of feedwater, both of the auxiliary feedwater trains were valved out of service.
2. The pressurizer electromatic relief valve, which opened during the initial pressure surge, failed to close when the pressure decreased below the actuation level.
3. Following rapid depressurization of the pressurizer, the pressurizer level indication may have lead to erroneous inferences of high level in the reactor coolant system. The pressurizer level indication apparently led the operators to prematurely terminate high pressure injection flow, even though substantial voids existed in the reactor coolant system.
4. Because the containment does not isolate on high pressure injection (HPI) initiation, the highly radioactive water from the relief valve discharge was pumped out of the containment by the automatic initiation of a transfer pump. This water entered the radioactive waste treatment system in the auxiliary building where some of it overflowed to the floor. Outgassing from this water and discharge through the auxiliary building ventilation system and filters was the principal source of the offsite release of radioactive noble gases.
5. Subsequently, the high pressure injection system was intermittently operated attempting to control primary coolant inventory losses through the electromatic relief valves apparently based on pressurizer level indication. Due to the presence of steam and/or noncondensable voids elsewhere in the reactor coolant system, this led to a further reduction in primary coolant inventory.

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6. Tripping of reactor coolant pumps during the course of the transient, to protect against pump damage due to pump vibration, led to fuel damage since voids in the reactor coolant system prevented natural circulation.

Actions To Be Taken by Licensees:

For all Babcock and Wilcox pressurized water reactor facilities with an operating license (the actions specified below replace those specified in IE Bulletin 79-05):

1. (This item clarifies and expands upon item 1. of IE Bulletin 79-05.)

In addition to the review of circumstances described in Enclosure 1 of IE Bulletin 79-05, review the enclosed preliminary chronology of the TMI-2 3/28/79 accident. This review should be directed toward understanding the sequence of events to ensure against such an accident at your facility(ies).

2. (This item clarifies and expands upon item 2. of IE Bulletin 79-05.)

Review any transients similar to the Davis Besse event (Enclosure 2 of IE Bulletin 79-05) and any others which contain similar elements from the enclosed chronology (Enclosure 1) which have occurred at your facility(ies). If any significant deviations from expected performance are identified in you review, provide details and an analysis of the safety significance together with a description of any corrective actions taken. Reference may be made to previous information provided to the NRC< if appropriate, in responding to this item.

3. (This item clarifies item 3. of IE Bulletin 79-05.)

Review the actions required by your operating procedures for coping with transients accident, with particular attention to:

- a. Recognition of the possibility of forming voids in the primary coolant system large enough to compromise the core cooling capability, especially natural circulation capability.
- b. Operator action required to prevent the formation of such voids.
- c. Operator action required to enhance core cooling in the event such voids are formed.

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4. (This item clarifies and expands upon item 4. of IE Bulletin 79-05.)

Review the actions directed by the operating procedures and training instructions to ensure that:

- a. Operators do not override automatic actions of engineered safety features.
- b. Operating procedures currently, or are revised to, specify that if the high pressure injection (HPI) system has been automatically actuated because of low pressure condition, it must remain in operation until either:
 - (1) Both low pressure injection (LPI) pumps are in operation and flowing at a rate in excess of 1000 gpm each and the situation has been stable for 20 minutes, or
 - (2) The HPI system has been in operation for 20 minutes, and all hot and cold leg temperatures are at least 50 degrees below the saturation temperature for the existing RCS pressure. If 50 degree subcooling cannot be maintained after HPI cutoff, the HPI shall be reactivated.
- c. Operating procedures currently, or are revised to, specify that in the event of HPI initiation, with reactor coolant pumps (RCP) operating, at least one RCP per loop shall remain operating.
- d. Operators are provided additional information and instructions to not rely upon pressurizer level indication alone, but to also examine pressurizer pressure and other plant parameter indications in evaluating plant conditions, e.g., water inventory in the reactor primary system.

5. (This item revises item 5. of IE Bulletin 79-05.)

Verify that emergency feedwater valves are in the open position in accordance with item 8 below. Also, review all safety-related valve positions and positioning requirements to assure that valves are positioned (open or closed) in a manner to ensure the proper operation of engineered safety features. Also review related procedures, such as those for maintenance and testing, to ensure that such valves are returned to their correct positions following necessary manipulations.

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- 6. Review the containment isolation initiation design and procedures, and prepare and implement all changes necessary to cause containment isolation of all lines whose isolation does not degrade core cooling capability upon automatic initiation of safety injection.
- 7. For manual valves or manually-operated motor-driven valves which could defeat or compromise the flow of auxiliary feedwater to the steam generators, prepare and implement procedures which:
 - a. require that such valves be locked in their correct position; or
 - b. require other similar positive position controls.

8. Prepare and implement immediately procedures which assure that two independent steam generator auxiliary feedwater flow paths, each with 100% flow capacity, are operable at any time when heat removal from the primary system is through the steam generators. When two independent 100% capacity flow paths are not available, the capacity shall be restored within 72 hours or the plant shall be placed in a cooling mode which does not rely on steam generators for cooling within the next 12 hours.

When at least one 100% capacity flow path is not available, the reactor shall be made subcritical within one hour and the facility placed in a shutdown cooling mode which does not rely on steam generators for cooling within 12 hours or at the maximum safe shutdown rate.

9. (This item revises item 6 of IE Bulletin 79-05.)

Review your operating modes and procedures for all systems designed to transfer potentially radioactive gases and liquids out of the primary containment to assure that undesired pumping of radioactive liquids and gases will not occur inadvertently.

In particular, ensure that such an occurrence would not be caused by the resetting of engineered safety features instrumentation. List all such systems and indicate:

- a. Whether interlocks exist to prevent transfer when high radiation indication exists, and
- b. Whether such systems are isolated by the containment isolation signal.

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10. Review and modify as necessary your maintenance and test procedures to ensure that they require:
 - a. Verification, by inspection, of the operability of redundant safety-related systems prior to the removal of any safety-related system from service.
 - b. Verification of the operability of all safety-related systems when they are returned to service following maintenance or testing.
 - c. A means of notifying involved reactor operating personnel whenever a safety-related system is removed from and returned to service.
11. All operating and maintenance personnel should be made aware of the extreme seriousness and consequences of the simultaneous blocking of both auxiliary feedwater trains at the Three Mile Island Unit 2 plant and other actions taken during the early phases of the accident.

12. Review your prompt reporting procedures for NRC notification to assure very early notification of serious events.

For Babcock and Wilcox pressurized water reactor facilities with an operating license, respond to Items 1, 2, 3, 4.a and 5 by April 11, 1979. Since these items are substantially the same as those specified in IE Bulletin 79-05, the required date for response has not been changed. Respond to Items 4.b through 4.d, and 6 through 12 by April 16, 1979.

Reports should be submitted to the Director of the appropriate NRC Regional Office and a copy should be forwarded to the NRC Office of Inspection and Enforcement, Division of Reactor Operations Inspection, Washington, DC 20555.

For all other reactors with an operating license or construction permit, this Bulletin is for information purposes and no written response is required.

Approved by GAO, B 180225 (R0072); clearance expires 7-31-80. Approval was given under a blanket clearance specifically for identified generic problems.

Enclosures:

1. Preliminary Chronology of TMI-2 3/38/79 Accident Until Core Cooling Restored.
2. List of IE Bulletins issued in last 12 months.

Enclosure 1 to
IE Bulletin 79-05A
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PRELIMINARY

CHRONOLOGY OF TMI-2 3/28/79 ACCIDENT UNTIL CORE COOLING RESTORED

TIME (Approximate)	EVENT
about 4 AM (t = 0)	Loss of Condensate Pump Loss of Feedwater Turbine Trip
t = 3-6 sec.	Electromatic relief valve opens (2255 psi) to relieve pressure in RCS
t = 9-12 sec.	Reactor trip on high RCS pressure (2355 psi)
t = 12-15 sec.	RCS pressure decays to 2205 psi (relief valve should have closed)
t = 15 sec.	RCS hot leg temperature peaks at 611 degrees F, 2147 psi (450 psi over saturation)

t = 30 sec. All three auxiliary feed water pumps running at pressure (Pumps 2A and 2B started at turbine trip). No flow was injected since discharge valves were closed.

t = 1 min. Pressurizer level indication begins to rise rapidly

t = 1 min. Steam Generators A and B secondary level very low - drying out over next couple of minutes.

t = 2 min. ECCS initiation (HPI) at 1600 psi

t = 4 - 11 min. Pressurizer level off scale - high - one HPI pump manually tripped at about 4 min. 30 sec. Second pump tripped at about 10 min. 30 sec.

t = 6 min. RCS flashes as pressure bottoms out at 1350 psig (Hot leg temperature of 584 degrees F)

t = 7 min., 30 sec. Reactor building sump pump came on.

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TIME EVENT

t = 8 min. Auxiliary feedwater flow is initiated by opening closed valves

t = 8 min. 21 sec. Steam Generator A pressure starts to recover

t = 11 min. Pressurizer level indication comes back on scale and decreases

t = 11-12 min. Makeup Pump (ECCS HPI flow) restarted by operators

t = 15 min. RC Drain/Quench Tank rupture disk blows at 190 psig (setpoint 200 psig) due to continued discharge of electromatic relief valve

t = 20 - 60 min. System parameters stabilized in saturated condition at about 1015 psig and about 550 degrees F.

t = 1 hour, 15 min. Operator trips RC pumps in Loop B

t = 1 hour, 40 min. Operator trips RC pumps in Loop A

t = 1-3/4 - 2 hours CORE BEGINS HEAT UP TRANSIENT - Hot leg temperature begins to rise to 620 degrees F (off scale within 14 minutes) and cold leg temperature drops to 150 degrees F. (HPI water)

t = 2.3 hour Electromatic relief valve isolated by operator
after S.G.-B isolated to prevent leakage

t = 3 hours RCS pressure increases to 2150 psi and
electromatic relief valve opened

t = 3.25 hours RC drain tank pressure spike of 5 psig

t = 3.8 hours RC drain tank pressure spike of 11 psi -RCS
pressure 1750; containment pressure increases
from 1 to 3 psig

t = 5 hour Peak containment pressure of 4.5 psig

t = 5 - 6 hours RCS pressure increased from 1250 psi to 2100
psi

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TIME EVENT

t = 7.5 hours Operator opens electromatic relief valve to
depressurize RCS to attempt initiation of RHR
at 400 psi

t = 8 - 9 hours RCS pressure decreases to about 500 psi Core
Flood Tanks partially discharge

t = 10 hour 28 psig containment pressure spike, containment
sprays initiated and stopped after 500 gal. of
NaOH injected (about 2 minutes of operation)

t = 13.5 hours Electromatic relief valve closed to
repressurize RCS, collapse voids, and start RC
pump

t = 13.5 - 16 hours RCS pressure increased from 650 psi to 2300 psi

t = 16 hours RC pump in Loop A started, hot leg temperature
decreases to 560 degrees F, and cold leg
temperature increases to 400 degrees F.
indicating flow through steam generator

Thereafter S/G "A" steaming to condensor
Condensor vacuum re-established
RCS cooled to about 280 degrees F., 1000 psi

Now (4/4) High radiation in containment
All core thermocouples less than 460 degrees
F. Using pressurizer vent valve with small
makeup flow

Slow cool down
RB pressure negative