

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, DC 20555

CASE

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 valve, 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 your 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 and accidents, 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.

264 124

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.

264 125

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.

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.

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 feedwater 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.

TIME	EVENT
t = 8 min.	Auxiliary feedwater flow is initiated by opening closed valves
t = 8 min. 18 sec.	Steam Generator B pressure reached minimum
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 hours	Peak containment pressure of 4.5 psig
t = 5 - 6 hours	RCS pressure increased from 1250 psi to 2100 psi

264 129

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 cooldown RB pressure negative

LISTING OF IE BULLETINS
ISSUED IN LAST TWELVE MONTHS

Bulletin No.	Subject	Date Issued	Issued To
78-05	Malfunctioning of Circuit Breaker Auxiliary Contact Mechanism-General Model CR105X	4/14/78	All Power Reactor Facilities with an OL or CP
78-06	Defective Cutler- Hammer, Type M Relays With DC Coils	5/31/78	All Power Reactor Facilities with an OL or CP
78-07	Protection afforded by Air-Line Respirators and Supplied-Air Hoods	6/12/78	All Power Reactor Facilities with an OL, all class E and F Research Reactors with an OL, all Fuel Cycle Facilities with an OL, and all Priority 1 Material Licensees
78-08	Radiation Levels from Fuel Element Transfer Tubes	6/12/78	All Power and Research Reactor Facilities with a Fuel Element transfer tube and an OL.
78-09	BWR Drywell Leakage Paths Associated with Inadequate Drywell Closures	6/14/79	All BWR Power Reactor Facilities with an OL or CP
78-10	Bergen-Paterson Hydraulic Shock Suppressor Accumulator Spring Coils	6/27/78	All BWR Power Reactor Facilities with an OL or CP

264 131

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ISSUED IN LAST TWELVE MONTHS

Bulletin No.	Subject	Date Issued	Issued To
78-11	Examination of Mark I Containment Torus Welds	7/21/78	BWR Power Reactor Facilities for action: Peach Bottom 2 and 3, Quad Cities 1 and 2, Hatch 1, Monticello and Vermont Yankee
78-12	Atypical Weld Material in Reactor Pressure Vessel Welds	9/29/78	All Power Reactor Facilities with an OL or CP
78-12A	Atypical Weld Material in Reactor Pressure Vessel Welds	11/24/78	All Power Reactor Facilities with an OL or CP
78-12B	Atypical Weld Material in Reactor Pressure Vessel Welds	3/19/79	All Power Reactor Facilities with an OL or CP
78-13	Failures In Source Heads of Kay-Ray, Inc., Gauges Models 7050, 7050B, 7051, 7051B, 7060, 7060B, 7061 and 7061B	10/27/78	All general and specific licensees with the subject Kay-Ray, Inc. gauges
78-14	Deterioration of Buna-N Components In ASCO Solenoids	12/19/78	All GE BWR facilities with an OL or CP
79-01	Environmental Qualification of Class IE Equipment	2/8/79	All Power Reactor Facilities with an OL or CP

LISTING OF IE BULLETINS
ISSUED IN LAST TWELVE MONTHS

Bulletin No.	Subject	Date Issued	Issued To
79-02	Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts	3/2/79	All Power Reactor Facilities with an OL or CP
79-03	Longitudinal Weld Defects In ASME SA-312 Type 304 Stainless Steel Pipe Spools Manufactured By Youngstown Welding and Engineering Co.	3/12/79	All Power Reactor Facilities with an OL or CP
79-04	Incorrect Weights for Swing Check Valves Manufactured by Velan Engineering Corporation	3/30/79	All Power Reactor Facilities with an OL or CP
79-05	Nuclear Incident at Three Mile Island	4/1/79	All B&W Power Reactor Facilities with an OL

264 133