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

REVIEW OF OPERATIONAL ERRORS AND SYSTEM MISALIGNMENTS IDENTIFIED DURING THE THREE MILE ISLAND INCIDENT

Description of Circumstances:

IE Bulletin 79-06 identified actions to be taken by the licensees of all pressurized water power reactors (except Babcock & Wilcox reactors) as a result of the Three Mile Island Unit 2 incident. This Bulletin clarifies the actions of Bulletin 79-06 for reactors designed by Combustion Engineering, and the response to this bulletin will eliminate the need to respond to Bulletin 79-06.

Actions to be taken by Licensees:

For all Combustion Engineering pressurized water reactor facilities with an operating license (the actions specified below replace those identified in IE Bulletin 79-06 on an item by item basis):

1. Review the description of circumstances described in Enclosure 1 of IE Bulletin 79-05 and the preliminary chronology of the TMI-2 3/28/79 accident included in Enclosure 1 to IE Bulletin 79-05A.

    a. This review should be directed toward understanding: (1) 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; (2) the apparent operational errors which led to the eventual core damage; (3) that the potential exists, under certain accident or transient conditions, to have a water level in the pressurizer simultaneously with the reactor vessel not full of water; and (4) the necessity to systematically analyze plant conditions and parameters and take appropriate corrective action.

    b. Operational personnel should be instructed to: (1) not override automatic action of engineered safety features unless continued operation of engineered safety features will result in unsafe plant conditions (see Section 6a.); and (2) not make operational decisions based solely on a single plant parameter indication when one or more confirmatory indications are available.

c. All licensed operators and plant management and supervisors with operational responsibilities shall participate in this review and such participation shall be documented in plant records.

2. 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. (e.g., remote venting)

3. Review the containment isolation initiation design and procedures, and prepare and implement all changes necessary to permit containment isolation whether manual or automatic, of all lines whose isolation does not degrade needed safety features or cooling capability, upon automatic initiation of safety injection.

4. For facilities for which the auxiliary feedwater system is not automatically initiated, prepare and implement immediately procedures which require the stationing of an individual (with no other assigned concurrent duties and in direct and continuous communication with the control room) to promptly initiate adequate auxiliary feedwater to the steam generator(s) for those transients or accidents the consequences of which can be limited by such action.

5. For your facilities, prepare and implement immediately procedures which:

   a. Identify those plant indications (such as valve discharge piping temperature, valve position indication, or valve discharge relief tank temperature or pressure indication) which plant operators may utilize to determine that pressurizer power operated relief valve(s) are open, and

   b. Direct the plant operators to manually close the power operated relief block valve(s) when reactor coolant system pressure is reduced to below the set point for normal automatic closure of the power operated relief valve(s) and the valve(s) remain stuck open.

6. Review the action directed by the operating procedures and training instructions to ensure that:

   a. Operators do not override automatic actions of engineered safety features, unless continued operation of engineered safety features will result in unsafe plant conditions. For example, if continued operation of engineered safety features would threaten reactor vessel integrity then the HPI should be secured (as noted in b(2) below).

   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 for 20 minutes or longer; at a rate which would assure stable plant behavior; 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 degress subcooling cannot be maintained after HPI cutoff, the HPI shall be reactivated. The degree of subcooling beyond 50 degrees F and the length of time HPI is in operation shall be limited by the pressure/temperature considerations for the vessel integrity.

   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 shall remain operating in each loop as long as the pump(s) is providing forced flow.

   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.

7. Review all safety-related valve positions, positioning requirements and positive controls to assure that valves remain 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, testing, plant and system startup, and supervisory periodic (e.g., daily/shift checks,) surveillance to ensure that such valves are returned to their correct positions following necessary manipulations and are maintained in their proper positions during all operational modes.

8. 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, venting or other release 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.

   c. The basis on which continued operability of the above features is assured.

9. Review and modify as necessary your maintenance and test procedures to ensure that they require:

   a. Verification, by test or 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. Explicit notification of involved reactor operational personnel whenever a safety-related system is removed from and returned to service.

10. Review your prompt reporting procedures for NRC notification to assure that NRC is notified within one hour of the time the reactor is not in a controlled or expected condition of operation. Further, at that time an open continuous communication channel shall be established and maintained with NRC.

11. Review operating modes and procedures to deal with significant amounts of hydrogen gas that may be generated during a transient or other accident that would either remain inside the primary system or be released to the containment.

12. Propose changes, as required, to those technical specifications which must be modified as a result of your implementing the above items.

For all light water reactor facilities designed by Combustion with an operating license, respond to Items 1-11 within 10 days of the receipt of this Bulletin. Respond to item 12 (Technical Specification Change proposals) in 30 days.

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, D.C. 20555.

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

Approved by GAO, B180225 (R0072); clearance expires 7/31/80. Approval was given under a blanket clearance specifically for identified generic problems.

Enclosure:
List of IE Bulletins Issued in Last
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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 Electric Model CR105X	4/14/78	All Power Reactor Facilities with an Operating License (OL) or Construction Permit (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 I Material Licensees
78-08	Radiation Levels from Fuel Element Transfer Tubes	6/12/78	All Power, Test and Research Reactor Facilities with an OL having Fuel Element Transfer Tubes
78-09	BWR Drywell Leakage Paths Associated with Inadequate Drywell Closures	6/14/78	All BWR Power Reactor Facilities with an OL (for action) or CP (for information)
78-10	Bergen-Paterson Hydraulic Shock Suppressor Accumulator Spring Coils	6/27/78	All BWR Power Reactor Facilities with an OL or CP

LISTING OF IE BULLETINS
ISSUED IN LAST TWELVE MONTHS (CONTINUED)

Bulletin No.	Subject	Date Issued	Issued To
78-11	Examination of Mark I Containment Torus Welds	7/24/78	BWR Power Reactor Facilities with an OL for action: Peach Bottom 2 and 3, Quad Cities 1 and 2, Hatch 1, Monticello and Vermont Yankee. All other BWR Power Reactor Facilities with an OL for information
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 (for action), and all other Power Reactor Facilities with an OL or CP (for information)

LISTING OF IE BULLETINS
ISSUED IN LAST TWELVE MONTHS (CONTINUED)

Bulletin No.	Subject	Date Issued	Issued to
79-01	Environmental Qualification of Class IE Equipment	2/8/79	All Power Reactor Facilities with an OL, except the 11 Systematic Evaluation Program Plants (for action), and all other Power Reactor Facilities with an OL or CP (for information)
79-02	Pipe Support Base Plate Design Using Concrete Expansion Anchor Bolts	3/8/79	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 Company	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 Babcock and Wilcox Power Reactor Facilities with an OL, Except Three Mile Island 1 and 2 (For Action), and All Other Power Reactor Facilities With an OL or CP (For Information)
79-05A	Nuclear Incident at Three Mile Island - Supplement	4/5/79	Same as 79-05

LISTING OF IE BULLETINS
ISSUED IN LAST TWELVE MONTHS (CONTINUED)

Bulletin No.	Subject	Date Issued	Issued to
79-06	Review of Operational Errors and System Mis-alignments Identified During the Three Mile Incident	4/11/79	All Pressurized Water Power Reactor Facilities with an OL Except B&W Facilities (For Action), All Other Power Reactor Facilities With an OL or CP (For Information)
79-06A	Same Title as 79-06	4/14/79	All Westinghouse Designed Pressurized Power Reactor Facil-ities with an OL (For Action), and All Other Power Reactor Facilities with an OL or CP (For Information)