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

APRIL 21, 1979

IE Bulletin 79-05B

NUCLEAR INCIDENT AT THREE MILE ISLAND - SUPPLEMENT

Description of Circumstances:

Continued NRC evaluation of the nuclear incident at Three Mile Island Unit 2 has identified measures in addition to those discussed in IE Bulletin 79-05 and 79-05A which should be acted upon by licensees with reactors designed by B&W. As discussed in Item 4.c. of Actions to be taken by Licensees in IEB 79-05A, the preferred mode of core cooling following a transient or accident is to provide forced flow using reactor coolant pumps.

It appears that natural circulation was not successfully achieved upon securing the reactor coolant pumps during the first two hours of the Three Mile Island (TMI) No. 2 incident of March 28, 1979. Initiation of natural circulation was inhibited by significant coolant voids, possibly aggravated by release of noncondensible gases, in the primary coolant system. To avoid this potential for interference with natural circulation, the operator should ensure that the primary system is subcooled, and remains subcooled, before any attempt is made to establish natural circulation.

Natural circulation in Babcock and Wilcox reactor systems is enhanced by maintaining a relatively high water level on the secondary side of the once through steam generators (OTSG). It is also promoted by injection of auxiliary feedwater at the upper nozzles in the OTSGs. The integrated Control System automatically sets the OTSG level setpoint to 50% on the operating range when all reactor coolant pumps (RCP) are secured. However, in unusual or abnormal situations, manual actions by the operator to increase steam generator level will enhance natural circulation capability in anticipation of a possible loss of operation of the reactor coolant pumps. As stated previously, forced flow of primary coolant through the core is preferred to natural circulation.

Other means of reducing the possibility of void formation in the reactor coolant system are:

A. Minimize the operation of the Power Operated Relief Valve (PORV) on the pressurizer and thereby reduce the possibly of pressure reduction by a blowdown through a PORV that was stuck open.

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B. Reduce the energy input to the reactor coolant system by a prompt reactor trip during transients that result in primary system pressure increases.

This bulletin addresses, among other things, the means to achieve these objectives.

Actions To Be Taken by Licensees:

For all Babcock and Wilcox pressurized water reactor facilities with an operating license: (Underlined sentences are modifications to, and supersede, IEB-79-05A).

1. Develop procedures and train operation personnel on methods of establishing and maintaining natural circulation. The procedures and training must include means of monitoring heat removal efficiency by available plant instrumentation. The procedures must also contain a method of assuring that the primary coolant system is subcooled by at least 50°F before natural circulation is initiated.

   In the event that these instructions incorporate anticipatory filling of the OTSG prior to securing the reactor coolant pumps, a detailed analysis should be done to provide guidance as to the expected system response. The instructions should include the following precautions:

   a. maintain pressurizer level sufficient to prevent loss of level indication in the pressurizer;

   b. assure availability of adequate capacity of pressurizer heaters, for pressure control and maintain primary system pressure to satisfy the subcooling criterion for natural circulation;

   c. maintain pressure - temperature envelope within Appendix G limits for vessel integrity.

   Procedures and training shall also be provided to maintain core cooling in the event both main feedwater and auxiliary feedwater are lost while in the natural circulation core cooling mode.

2. Modify the actions required in Items 4a and 4b of IE Bulletin 79-05A to take into account vessel integrity considerations.

   "4. 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, <u>unless continued operation of engineered</u>

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 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 degrees 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."

3. Following detailed analysis, describe the modifications to design and procedures which you have implemented to assure the reduction of the likelihood of automatic actuation of the pressurizer PORV during anticipated transients. This analysis shall include consideration of a modification of the high pressure scram setpoint and the PORV opening setpoint such that reactor scram will preclude opening of the PORV for the spectrum of anticipated transients discussed by B&W in Enclosure 1. Changes developed by this analysis shall not result in increased frequency of pressurizer safety valve operation for these anticipated transients.

4. Provide procedures and training to operating personnel for a prompt manual trip of the reactor for transients that result in a pressure increase in the reactor coolant system. These transients include:

a. Loss of main feedwater

b. Turbine trip

c. Main Steam Isolation Valve closure

d. Loss of offsite power

e. Low OTSG level

f. Low pressurizer level

5. Provide for NRC approval a design review and schedule for implementation of a safety grade automatic anticipatory reactor scram for loss of feed-water, turbine trip, or significant reduction in steam generator level.

6. The actions required in item 12 of IE Bulletin 79-05A are modified as follows:

   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.

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

Response schedule for B&W designed facilities:

a. For Items 1, 2, 4 and 6, all facilities with an operating license respond within 14 days of receipt of this Bulletin.

b. For Item 3, all facilities currently operating, respond within 24 hours. All facilities with an operating license, not currently operating, respond before resuming operation.

c. For Items 5 and 7, all facilities with an operating license respond 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, B180255 (R0072); clearance expires 7/31/80. Approval was given under a blanket clearance specifically for identified generic problems.

INTRODUCTION

THE CONTINUING REVIEW OF THE SEQUENCE OF EVENTS LEADING TO THE INCIDENT AT TMI-2 ON MARCH 28, 1979 SHOWS THAT ACTION CAN BE TAKEN TO PROVIDE ASSURANCE THAT THE PILOT-OPERATED RELIEF VALVE (PORV) MOUNTED ON THE PRESSURIZER OF B&W PLANTS WILL NOT BE ACTUATED BY ANTICIPATED TRANSIENTS WHICH HAVE OCCURRED OR HAVE A SIGNIFICANT PROBABILITY OF OCCURRING IN THESE PLANTS. THIS ACTION MUST NOT DEGRADE THE SAFETY OF THE AFFECTED PLANTS WITH RESPECT TO THEIR RESPONSE TO NORMAL, UPSET OR ACCIDENT CONDITIONS NOR LEAD TO UNREVIEWED SAFETY CONCERNS. THE ANTICIPATED TRANSIENTS OF CONCERN ARE:

1. LOSS OF EXTERNAL ELECTRICAL LOAD
2. TURBINE TRIP
3. LOSS OF MAIN FEEDWATER
4. LOSS OF CONDENSER VACUUM
5. INADVERTENT CLOSURE OF MAIN STEAM ISOLATION VALVES (MSIV).

A NUMBER OF ALTERNATIVES WERE CONSIDERED IN DEVELOPING THE ACTIONS PROPOSED BELOW INCLUDING:

1. RESTRICTING REACTOR POWER TO A VALUE WHICH WOULD ASSURE NO ACTUATION OF THE PORV. THE REACTOR PROTECTION SYSTEM, DESIGN PRESSURE AND PORV SETPOINTS REMAINED AT THEIR CURRENT VALUES.

2. LOWERING THE HIGH PRESSURE REACTOR TRIP SETPOINT TO A VALUE WHICH WOULD ASSURE NO ACTUATION OF THE PORV. THE DESIGN PRESSURE OF THE REACTOR AND THE SETPOINT FOR PORV ACTUATION REMAINED AT THEIR CURRENT VALUES.

3. LOWERING THE HIGH PRESSURE REACTOR TRIP SETPOINT AND ADJUSTING THE OPERATING PRESSURE (AND TEMPERATURE) OF THE REACTOR TO ASSURE NO PORV ACTUATION AND TO PROVIDE ADEQUATE MARGIN TO ACCOMMODATE VARIATIONS IN OPERATING PRESSURE. THE SETPOINT FOR PORV ACTUATION REMAINED AT ITS CURRENT VALUE. THIS ALTERNATIVE WOULD REDUCE NET ELECTRICAL OUTPUT.

4. ADJUSTING THE HIGH PRESSURE TRIP AND THE PORV SETPOINTS TO ASSURE NO PORV ACTUATION FOR THE CLASS OF ANTICIPATED EVENTS OF CONCERN. THE DESIGN PRESSURE OF THE REACTOR REMAINED AT ITS CURRENT VALUE.

AN ANALYSIS OF THE IMPACT OF THESE VARIOUS ALTERNATIVES AND THEIR CONTRIBUTION TO ASSURING THAT THE PORV WILL NOT ACTUATE FOR THE CLASS OF ANTICIPATED TRANSIENTS OF CONCERN HAS BEEN COMPLETED. THE RESULTS SHOW THAT:

LOWERING THE HIGH PRESSURE REACTOR TRIP SETPOINT FROM <u>2355 PSIG TO 2300 PSIG</u>

AND

RAISING THE SETPOINT FOR THE PILOT OPERATED RELIEF VALVE FROM <u>2255 PSIG TO 2450 PSIG</u>

PROVIDES THE REQUIRED ASSURANCE. THIS ACTION HAS THE FURTHER ADVANTAGES OF:

EXTRACT OF B&W COMMUNICATION - RECEIVED BY NRC
4/20/79

1. REDUCING THE PROBABILITY OF PORV AND ASME CODE PRESSURIZER SAFETY VALVE ACTUATION FOR OTHER INCREASING PRESSURE TRANSIENTS.
2. PRESERVING PRESSURE RELIEF CAPACITY FOR ALL HIGH PRESSURE TRANSIENTS.
3. ELIMINATING THE POSSIBILITY OF INTRODUCING UNREVIEWED SAFETY CONCERNS.
4. REDUCING THE TIME AT WHICH THE STEAM SYSTEM HEAT SINK WOULD BE LOST IN THE EVENT EMERGENCY FEEDWATER FLOW WERE DELAYED.

A SUMMARY OF THE IMPACT OF THE PROPOSED SETPOINT CHANGES ON ALL ANTICIPATED TRANSIENTS IS GIVEN IN TABLE 1.

B&W PLANTS ARE CURRENTLY CAPABLE OF RUNBACK TO 15% OF FULL POWER UPON LOSS OF LOAD OR TRIP OF THE TURBINE. THIS CAPABILITY REQUIRES ACTUATION OF THE PILOT-OPERATED RELIEF VALVES. THE CAPABILITY INCREASES THE RELIABILITY OF POWER SUPPLY TO THE SYSTEM BY RETURNING THE UNITS TO POWER GENERATION MORE QUICKLY AFTER THESE TRANSIENTS. THE ACTION PROPOSED ABOVE WILL REQUIRE THAT THE REACTOR BE TRIPPED FOR THESE EVENTS.

NRC NOTE:

The effect of changing the reactor coolant system pressure trip setpoint upon peak pressurizer pressure is typified by the attached figure 1. which was developed by B&W for a loss of feedwater transient.

TABLE 1

SUMMARY OF PROTECTION AGAINST PORV ACTUATION PROVIDED BY PROPOSED SETPOINT CHANGES FOR ALL ANTICIPATED TRANSIENTS

EXTRACT OF B&W COMMUNICATION - RECEIVED BY NRC 4/20/79

1. ANTICIPATED TRANSIENTS WHICH HAVE OCCURRED AT B&W PLANTS AND WHICH WOULD NORMALLY ACTIVATE PORV AT THE <u>CURRENT</u> SETPOINT (2255 PSIG):

   A. TURBINE TRIP
   B. LOSS OF EXTERNAL ELECTRICAL LOAD
   C. LOSS OF MAIN FEEDWATER
   D. LOSS OF CONDENSER VACUUM
   E. INADVERTENT CLOSURE OF MSIV

2. ANTICIPATED TRANSIENTS WHICH HAVE OCCURRED AT B&W PLANTS AND WHICH WOULD NORMALLY ACTUATE PORV AT THE <u>PROPOSED</u> SETPOINT (2450 PSIG):

   NONE

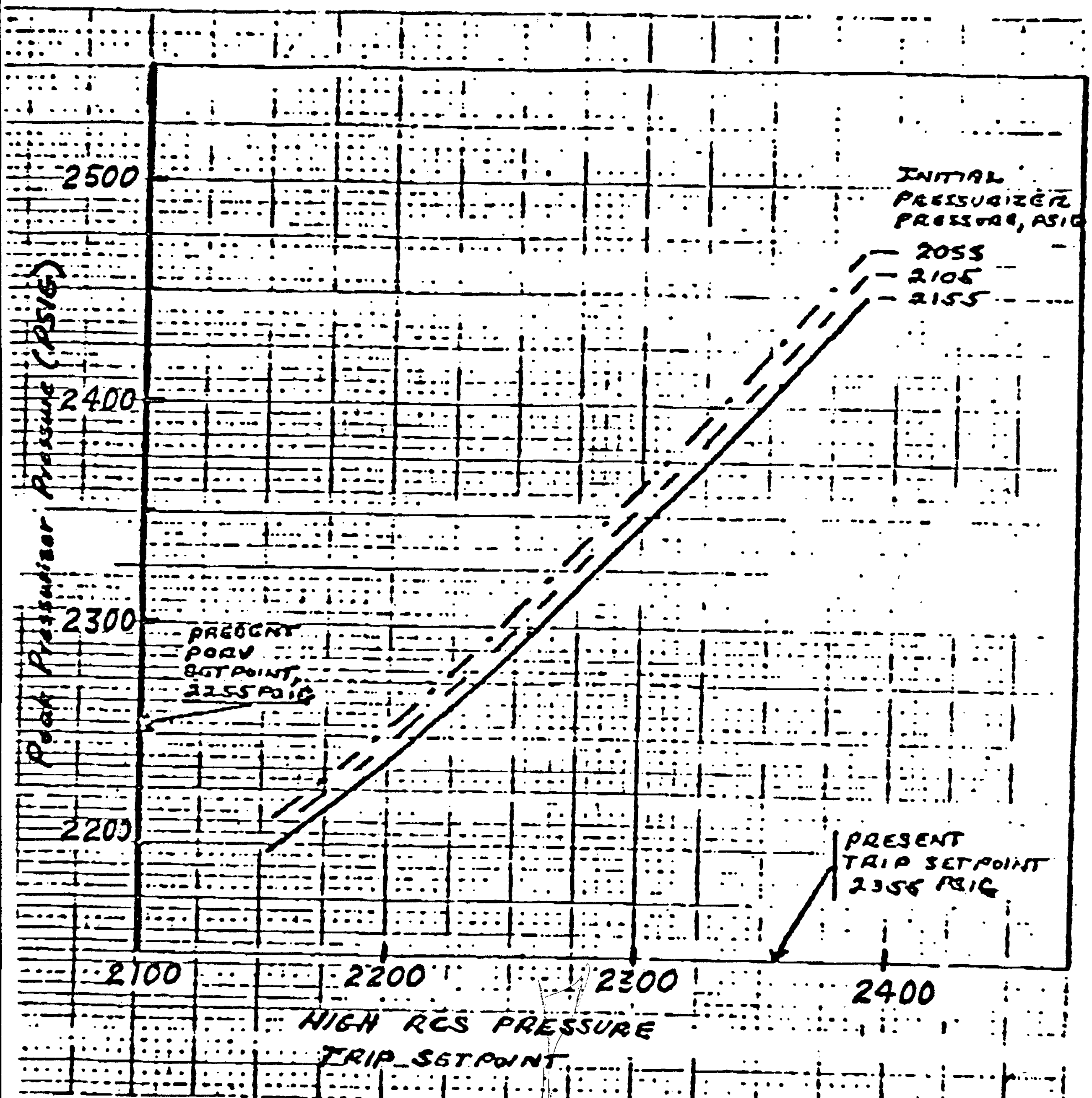
3. ANTICIPATED TRANSIENTS WHICH HAVE NOT OCCURRED AT B&W PLANTS (LOW PROBABILITY EVENTS) AND WHICH WOULD NORMALLY ACTUATE PORV AT THE <u>CURRENT</u> SETPOINT (2255 PSIG):

   A. SOME CONTROL ROD GROUP WITHDRAWALS (MODERATE TO HIGH REACTIVITY WORTH GROUPS NOT OTHERWISE PROTECTED BY HIGH FLUX TRIP).
   B. MODERATOR DILUTION.

4. ANTICIPATED TRANSIENTS WHICH HAVE NOT OCCURRED AT B&W PLANTS (LOW PROBABILITY EVENTS) AND WHICH WOULD ACTUATE THE PORV AT THE <u>PROPOSED</u> SETPOINT (2450 PSIG):

   A. SOME CONTROL ROD GROUP WITHDRAWALS (HIGH REACTIVITY WORTH NOT OTHERWISE PROTECTED BY HIGH FLUX TRIP).

EXTRACT OF B&W COMMUNICATION - RECEIVED BY NRC
4/20/79



Peak pressurizer pressure as a function of RCS pressure trip setpoint for a loss of feedwater transient for expected conditions and various initial pressures.

Figure 1

LISTING OF IE BULLETINS
ISSUED IN LAST TWELVE MONTHS

Bulletin No.	Subject	Date Issued	Issued To
79-06A Rev. 1	Review of Operational Errors and System Misalignments Identified During the Three Mile Island Incident	4/18/79	All Pressurized Water Power Reactor Facilities of Westinghouse Design with an OL
79-09	Failure of GE Type AK-2 Circuit Breaker in Safety Related Systems	4/17/79	All Power Reactor Facilities with an OL
79-08	Events Relevant to BWR Reactors Identified During Three Mile Island Incident	4/14/79	All BWR Power Reactor Facilities with an OL
79-07	Seismic Stress Analysis of Safety-Related Piping	4/14/79	All Power Reactor Facilities with an OL or CP
79-06B	Review of Operational Errors and System Misalignments Identified During the Three Mile Island Incident	4/14/79	All Combustion Engineering Designed Pressurized Water Power Reactor Facilities with an Operating License
79-06A	Review of Operational Errors and System Misalignments Identified During the Three Mile Island Incident	4/14/79	All Pressurized Water Power Reactor Facilities of Westinghouse Design with an OL
79-06	Review of Operational Errors and System Misalignments Identified During the Three Mile Island Incident	4/11/79	All Pressurized Water Power Reactors with an OL except B&W facilities
79-05A	Nuclear Indicent at Three Mile Island	4/5/79	All B&W Power Reactor Facilities with an OL
79-05	Nuclear Incident at Three Mile Island	4/2/79	All Power Reactor Facilities with an OL and CP

LISTING OF IE BULLETINS
ISSUED IN LAST TWELVE MONTHS

Bulletin No.	Subject	Date Issued	Issued To
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
79-03	Longitudinal Weld Defects in ASME SA-312 Type 3C4 Stainless Steel Pipe Spools Manufactured by Youngstown Welding and Engineering Co.	3/12/79	All Power Reactor Facilities with an OL or CP
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-C1	Environmental Qualification of Class IE Equipment	2/8/79	All Power Reactor Facilities with an OL or CP
78-14	Deterioration of BUNA-N Components in ASCO	12/19/78	All GE BWR facilities with an OL or CP
78-13	Failures In Source Heads of Kay-Ray, Inc., Gauges Models 7950, 7050B, 7051, 7051B, 7060, 7060B, 7061 and 7061B	10/27/78	All general and specific licensees with the subject Kay-Ray, Inc. gauges
78-12B	Atypical Weld Material in Reactor Pressure Vessel Welds	3/19/79	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
78-12	Atypical Weld Material in Reactor Pressure Vessel Welds	9/29/78	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
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
78-10	Bergen-Paterson Hydraulic Shock Suppressor Accumulator Spring Coils	6/27/78	All BWR Power Reactor Facilities with an OL or CP
78-09	BWR Drywell Leakage Paths Associated with Inadequate Drywell Closures	6/14/79	All BWR Power Reactor Facilities with an OL or CP
78-08	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
78-07	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
78-06	Defective Cutler-Hammer Type M Relays with DC Coils	5/31/78	All Power Reactor Facilities with an OL or CP

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

April 20, 1979

Honorable Victor Gilinsky
Acting Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Dr. Gilinsky:

This letter is in response to yours of April 18, 1979 which requested that the ACRS notify the Commissioners immediately if we believe any of our oral recommendations of April 17 should be acted upon before our next regularly scheduled meeting at which we could prepare a formal letter. The Committee discussed this topic by conference telephone call on April 19 and offers the following comments.

All of the recommendations made by the ACRS in its meeting with the Commissioners on April 17, 1979, are generic in nature and apply to all PWRs. None were intended to require immediate changes in operating procedures or plant modifications of operating PWRs. Such changes should be made only after study of their effects on overall safety. Such studies should be made by the licensees and their suppliers or consultants and by the NRC Staff. The Committee believes that these studies should be begun in the near future on a time scale that will not divert the NRC Staff or the industry representatives from their tasks relating to the cooldown of Three Mile Island Unit 2. However, the Committee believes that it would be possible and desirable to initiate immediately a survey of operating procedures for achieving natural circulation, including the case when offsite power is lost, and the role of the pressurizer heaters in such procedures.

At its meeting on April 16 and 17, 1979, the Committee discussed with the NRC Staff the matter of natural circulation for the Three Mile Island Unit 2 plant. The Committee believes that this matter is receiving careful attention by the NRC Staff and the licensee.

To EDO for Appropriate Action. Distribution: Chm, Cmrs, PE, OGC, OCA, SECY, PDR, OIA. Rapifaxed to EDO, PA, E. Case. 79-1117.

The Committee's own recommendations to the Commission on April 17 were not intended to apply to Three Mile Island Unit 2.

We plan to write a further report on these matters at our May 10, 1979 meeting.

Sincerely,

Max W. Carbon
Chairman

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

April 18, 1979

MEMORANDUM FOR: Chairman Hendrie
Commissioner Gilinsky
Commissioner Kennedy
Commissioner Bradford
Commissioner Ahearne

FROM: R. F. Fraley, Executive Director
Advisory Committee on Reactor Safeguards

Attached for your information and use is a copy of the recommendations of the Advisory Committee on Reactor Safeguards which were orally presented to and discussed with you on April 17, 1979 regarding the recent accident at the Three Mile Island Nuclear Station Unit 2.

R F Fraley
R. F. Fraley
Executive Director

Attachment:
Recommendations of the NRC Advisory Committee on Reactor Safeguards Re. the 3/28/79 Accident at The Three Mile Island Nuclear Station Unit 2

April 17, 1979

RECOMMENDATIONS OF THE NUCLEAR REGULATORY COMMISSION ADVISORY COMMITTEE ON REACTOR SAFEGUARDS REGARDING THE MARCH 28, 1979 ACCIDENT AT THE THREE MILE ISLAND NUCLEAR STATION UNIT 2

Presented orally to, and discussed with, the NRC Commissioners during the ACRS-Commissioners Meeting on April 17, 1979 - Washington, D. C.

Natural circulation is an important mode of reactor cooling, both as a planned process and as a process that may be used under abnormal circumstances. The Committee believes that greater understanding of this mode of cooling is required and that detailed analyses should be developed by licensees or their suppliers. The analyses should be supported, as necessary, by experiment. Procedures should be developed for initiating natural circulation in a safe manner and for providing the operator with assurance that circulation has, in fact, been established. This may require installation of instrumentation to measure or indicate flow at low water velocity.

The use of natural circulation for decay heat removal following a loss of offsite power sources requires the maintenance of a suitable overpressure on the reactor coolant system. This overpressure may be assured by placing the pressurizer heaters on a qualified onsite power source with a suitable arrangement of heaters and power distribution to provide redundant capability. Presently operating PWR plants should be surveyed expeditiously to determine whether such arrangements can be provided to assure this aspect of natural circulation capability.

The plant operator should be adequately informed at all times concerning the conditions of reactor coolant system operation which might affect the capability to place the system in the natural circulation mode of operation or to sustain such a mode. Of particular importance is that information which might indicate that the reactor coolant system is approaching the saturation pressure corresponding to the core exit temperature. This impending loss of system overpressure will signal to the operator a possible loss of natural circulation capability. Such a warning may be derived from pressurizer pressure instruments and hot leg temperatures in conjunction with conventional steam tables. A suitable display of this information should be provided to the plant operator at all times. In addition, consideration should be given to the use of the flow exit temperatures from the fuel subassemblies, where available, as an additional indication of natural circulation.

The exit temperature of coolant from the core is currently measured by thermocouples in many PWRs to determine core performance. The Committee recommends that these temperature measurements, as currently available, be used to guide the operator concerning core status. The range of the information displayed and recorded should include the full capability of the thermocouples. It is also recommended that other existing instrumentation be examined for its possible use in assisting operating action during a transient.

The ACRS recommends that operating power reactors be given priority with regard to the definition and implementation of instrumentation which provides additional information to help diagnose and follow the course of a serious accident. This should include improved sampling procedures under accident conditions and techniques to help provide improved guidance to offsite authorities, should this be needed. The Committee recommends that a phased implementation approach be employed so that techniques can be adopted shortly after they are judged to be appropriate.

The ACRS recommends that a high priority be placed on the development and implementation of safety research on the behavior of light water reactors during anomalous transients. The NRC may find it appropriate to develop a capability to simulate a wide range of postulated transient and accident conditions in order to gain increased insight into measures which can be taken to improve reactor safety. The ACRS wishes to reiterate its previous recommendations that a high priority be given to research to improve reactor safety.

Consideration should be given to the desirability of additional equipment status monitoring on various engineered safeguards features and their supporting services to help assure their availability at all times.

The ACRS is continuing its review of the implications of this accident and hope to provide further advice as it is developed.