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 Licenses 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 septoint 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 possibility 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 50F 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 Item 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, unless continued operation of engineered

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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 HP 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 rector 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.

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5. Provide for NRC approval a design review and schedule for

implementation of a safety grade automatic anticipatory reactor scram for loss of feedwater, 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 nd 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, B180225 (R0072); clearance expires 7/31/80. Approval was given under a blanket clearance specifically for identified generic problems.

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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
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

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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 timers. 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.

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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 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.