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

(Also ACRS-R-0825)

April 7, 1979

50-320

Honorable Joseph M. Hendrie
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: INTERIM REPORT ON RECENT ACCIDENT AT THE THREE MILE ISLAND NUCLEAR STATION UNIT 2

Dear Dr. Hendrie

During its 228th meeting, April 5-7, 1979, the Advisory Committee on Reactor Safeguards reviewed the circumstances relating to the recent accident at the Three Mile Island Nuclear Station Unit 2. During this review, the Committee had the benefit of discussions with the NRC Staff.

Our study of the accident at Three Mile Island has shown that it is very difficult for a PWR plant operator to understand and properly control the course of an accident involving a small break in the reactor coolant system accompanied by other abnormal conditions.

The Committee recommends that further analyses be made, as soon as possible, of transients and accidents in PWRs that involve initially, or at some time during their course, a small break in the primary system. The computer codes used for these analyses should be capable of predicting the conditions observed during the accident at Three Mile Island, including thermal-hydraulic effects and clad and fuel temperatures. The range of break sizes considered should include the smallest that could be deemed significant, and should consider a range of break locations.

The Committee believes that the analyses recommended above will demonstrate, as has the accident at Three Mile Island, that additional information regarding the status of the system will be needed in order for the plant operator to follow the course of an accident and thus be able to respond in an appropriate manner. As a minimum, and in the interim, it would be prudent to consider expeditiously the provision

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of instrumentation that will provide an unambiguous indication of the level of fluid in the reactor vessel. Early consideration should be given also to providing remotely controlled means for venting high points in the reactor system, as practical.

The foregoing recommendations apply to all pressurized water reactors.

The recommendations in IE Bulletin 79-05A, dated April 5, 1979, are believed to be generally suitable for Babcock and Wilcox facilities, on an interim basis. However, the Committee believes that the actions listed in Item 4b. under the heading, "Actions To Be Taken by Licensees," may prove to be unduly prescriptive in view of the uncertainties in predicting the course of anomalous transients or accidents involving small breaks in the primary system.

With regard to Three Mile Island Unit 2, the Committee believes that decisions should be made expeditiously with regard to contingency measures which may be prudent concerning containment and reactor cooldown as a backup to the currently planned cooldown procedure.

The Committee is continuing its review of these and other concerns arising from this accident and will provide further advice as it is developed.

Sincerely,

Max W Carbon

Max W. Carbon
Chairman

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