UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR REACTOR REGULATION
HAROLD R. DENTON, DIRECTOR

In the Matter of	)	
	)	
PETITION TO SUSPEND ALL OPERATING LICENSES FOR PRESSURIZED WATER REACTORS	)	Emergency Core Cooling Systems

DIRECTOR'S DECISION UNDER 10 C.F.R. 2.206

By petition dated March 29, 1979 the Environmental Coalition on Nuclear Power (ECNP) requested that the Nuclear Regulatory Commission (NRC) suspend all operating licenses for pressurized water reactors (PWRs). This petition has been considered under the provisions of 10 C.F.R. 2.206 of the Commission's regulations. Notice of receipt of the petition was published in the Federal Register December 6, 1979 (44 FR 70241).

The petition contends that safety evaluations for all operating PWRs are invalid and thus licenses for all PWRs should be suspended or revoked. Petitioner asserts that the consequences of the accident at Three Mile Island Unit 2 (TMI-2), at least some of the fuel melted, was in excess of the performance required for the Emergency Core Cooling System (ECCS) under 10 C.F.R. 50.46. Yet, the petitioner also contends, the accident which initiated the TMI-2 fuel damage was less severe than accidents specifically analyzed to demonstrate acceptable performance by the ECCS. Thus, petitioner contends, the analyses used to predict performance under the provisions of 10 C.F.R. 50.46 must be invalid and hence the basis for granting all PWRs licenses is invalid and these licenses should be suspended or revoked.

I have reviewed the information submitted by the Environmental Coalition on Nuclear Power and the issues addressed in the petition. For the reasons set forth below, petitioner's request that all operating licenses for PWRs should be suspended or revoked is denied.

Section 50.46 of the Commission's regulations requires that each boiling and pressurized light water nuclear power reactor must be provided with an emergency core cooling system designed in such a way that its calculated cooling performance following postulated loss-of-coolant accidents conforms to a set of criteria. Included in that set of criteria [10 C.F.R. 50.46(b)] is a requirement that the calculated maximum fuel element cladding temperature shall not exceed 2200°F. 10 C.F.R. 50.46 further requires that ECCS cooling performance is to be calculated: 1) in accordance with an acceptable evaluation model and 2) for a number of postulated loss-of-coolant accidents sufficient to provide assurance that the entire spectrum of loss-of-coolant accidents is covered. The spectrum of accidents examined includes a break equivalent in size to the double-ended rupture of the largest pipe of the reactor coolant system. (10 C.F.R. Part 50, Appendix K, I.C.1.)

On March 28, 1979, TMI-2 experienced a feedwater transient that, through a particular sequence of failures, led to a small break loss-of-coolant accident and resulted in significant core damage. The failures that were experienced occurred in the general areas of design, equipment malfunction, and human performance. This TMI-2 sequence of events and failures had not been previously analyzed and the fuel damage was beyond that predicted by 10 C.F.R. 50.46 analyses. Therefore, a question could be raised as to whether the analyses performed to meet 10 C.F.R.

50.46 were adequate, specifically: 1) whether the evaluation model used for compliance with 50.46 to evaluate the behavior of the reactor system during a postulated loss-of-coolant accident was adequate; and 2) whether there is sufficient assurance that a proper set of loss-of-coolant accidents has been analyzed to determine that the ECCS will perform as required.

In the NRC's Office of Inspection and Enforcement investigation of the TMI-2 accident (NUREG-0600, "Investigation into the March 28, 1979 Three Mile Island Accident by the Office of Inspection and Enforcement") it was stated that the TMI-2 accident could have been prevented in spite of any known or postulated inadequacies in transient and accident analyses. The forward to NUREG-0600 states:

> "The design of the plant, the equipment that was installed, the various accident and transient analyses, and the emergency procedures were adequate to have prevented the serious consequences of the accident, if they had been permitted to function or be carried out as planned. For example, had the operators allowed the emergency core cooling system to perform its intended function, damage to the core would most likely have been prevented."

NUREG-0600 estimates that during the initial 3½ hours of the accident the average ECCS flow was only about 25 gpm, because the operators had reduced the flow. As part of the TMI Inquiry, Battelle Columbus Laboratories explored alternative accident sequences. (NUREG/CR-1219, "Analysis of the Three Mile Island Accident and Alternative Sequences"). They concluded that if the high pressure injection (HPI) ECCS flow had not been throttled, full ECCS flow through the pumps would have remained above 800 gpm. As a result, the core would have remained covered, the fuel cladding temperature would not have increased at all, and the requirements of 10 CFR 50.46 would not have been violated.

Babcock and Wilcox has analyzed small break accidents similar in size to the TMI stuck open PORV. For these analyses ("Evaluation of Transient Behavior and Small Reactor Coolant System Breaks in the 177 Fuel Assembly Plant", May 7, 1979) B&W used methods which comply with the requirements of 50.46 and Appendix K to 10CFR50. The required single failure for these small breaks would mean that one of the HPI pumps did not work. The B&W analyses showed that the core remained covered for these small breaks even with half the total possible HPI flow. Both the BMI and B&W analyses were benchmarked successfully against the reduced HPI flow data from the TMI accident.

Clearly the TMI ECCS system was designed to cope with a TMI type accident, and would have, had the system not been overridden by the operators. 10 CFR 50.46 and Appendix K require that the ECCS be capable of mitigating the effects of an accident, assuming the most limiting single failure. However, it is recognized that the occurrence of multiple equipment failures and/or operator errors could result in conditions which exceed the core thermal limits of Appendix K. Such was the case at TMI. Contrary to the petioners contention, the events which occurred at TMI removed the plant from its design envelope, and placed it in a more severe condition than that required to be analyzed by Appendix K and 50.46. It is not reasonable to require protection from the effects of every conceivable combination of errors which could occur, without limiting the number of errors, because the number of such combinations is limitless.

One of the tasks of the NRR Bulletins and Order Task Force (B&OTF) formed in May 1979 was to generically evaluate feedwater transients, small breaks LOCAs, and other TMI-2 related events in operating plants to confirm or establish the basis for their continued safe operation. In order to fulfill this charter, B&OTF investigated a large spectrum of small breaks and transients to assure that the installed system for all modern operating light water reactors could adequately cope with these events. Reactor vendors, NRC consultants, and the NRC staff were required to analyze hundreds of cases in pursuit of this goal. As a result of this review, some parts of the analytical models were targeted for future review and possible improvement. However none of these sub-models were judged to have a substantial impact on ECCS system design. The analysis also aided in assessing operator guidelines for recognition and mitigation of small break LOCA (see NUREG-0645, " Report of the Bulletins and Orders Task Force" January 1980). Another major charter of B&OTF was to assure that all licensees were well trained in the recognition and mitigation of small break LOCAs and would not prematurely throttle or terminate the ECCS during such an event.

To implement the recommendations of all the internal and external TMI inquiries, a plan of action was devised (NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," U.S. Nuclear Regulatory Commission, published May 1980 and revised August 1980) in the form of a set of findings and requirements for safe operation of all reactors. These findings and requirements have been further elaborated in NUREG-0737, "Clarification of TMI Action Plan Requirements," U.S. Nuclear Regulatory Commission, November 1980. Included in this action plan is the small break model re-assessment. This activity has already begun and includes the comparison of affected analytical models to a large variety of experiments. To date the ability of these analysis tools has been encouraging. I see no evidence that they are not up to the task. The NRR staff constantly encourages small break evaluation model holders to make improvements.

In view of the above actions taken since the TMI-2 accident, I find no basis to conclude either from the assertions in the subject petition or from our current knowledge of loss-of-coolant accident analysis methods, that the analyses performed in compliance with 10 C.F.R. 50.46 are not valid.

In addition, as part of the TMI-Action Plan Requirements, a program to evaluate the uncertainties which may exist in small-break ECCS performance calculations has been proposed. Holders of approved ECCS evaluation models will evaluate these uncertainties; the Office of Nuclear Reactor Regulation will evaluate their results. If changes are needed in the analysis methods to properly account for these uncertainties, recommendations will be made to the Commission to adopt such changes. (See NUREG-0660, Task II.E.2)

On the basis of my conclusion that the analyses performed in compliance with 10 C.F.R. 50.46 are valid and in view of the many changes which have been imposed on PWRs, I find that continued operation of PWR's poses no undue risk to the public health and safety.

CONCLUSION

Based on the foregoing discussion and the provisions of 10 C.F.R. 2.206, I have determined that there are no adequate bases for suspension of PWR operating licenses. The request by the Environmental Coalition on Nuclear Power is, therefore, denied.

A copy of this decision will be placed in the Commission's Public Document Room at 1717 H Street, N. W., Washington, D. C. 20555 and each local public document room for all PWRs. A copy of this decision will also be filed with the Secretary for the Commission's review in accordance with 10 C.F.R. 2.206(c), of the Commission's regulations.

As provided in 10 C.F.R. 2.206(c), this decision will constitute the final action of the Commission twenty-five (25) days after the date of issuance, unless the Commission, on its own motion, institutes a review of this decision within that time.

Harold R. Denton
Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Dated at Bethesda Maryland
this 29th day of May, 1981.