UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of METROPOLITAN EDISON COMPANY, ET AL. (Three Mile Island Nuclear Station, Unit No. 1)	)))))	Docket No. 50-289

NRC STAFF TESTIMONY OF WALTON L. JENSEN, JR.

RELATIVE TO BOARD QUESTION REGARDING UCS CONTENTION 8

Board question regarding UCS Contention 8:

> The board directs the staff and the licensee to present experts and the fundamental documents involved in the small break LOCA analysis, and to have very complete testimony on this subject. The recommendations of NUREG-0565 and NUREG-0623 should be addressed.
>
> It appears from the small break LOCA analysis that there is a large amount of reliance upon operator action and on non-safety-grade equipment. The board wants that issue explored by testimony, including why such reliance is proper. Tr. 2374-85.

Response:

Testimony on the subject of small break LOCA analyses and supporting fundamental documents have been filed by the NRC in response to UCS Contention 8. This testimony did not specifically address the recommendations of NUREG-0565, "Generic Evaluation of Small Break Loss-of-Coolant Accident Behavior in Babcock & Wilcox Designed 177-FA Operating Plants," January 1980 or those of NUREG-0623, "Generic Assessment of Delayed Pump Trip During Small Break Loss-of-Coolant Accident in Pressurized Water Reactors," November 1979. With one exception, implementation of the recommendations relating to small break LOCA analysis

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were not required by the order for TMI-1 restart. The exception is the requirement of NUREG-0623 for tripping the reactor coolant pumps which has been implemented at TMI-1 as discussed below. All NUREG-0565 and NUREG-0623 recommendations will be implemented as part of the "NRC Action Plan developed as a result of the TMI-2 accident," NUREG-0660, Task II.K Item 3. A letter from D. Eisenhut, NRC, to All Licensees of Operating Plants and Applicants for Operating Licenses and Holders of Construction Permits dated September 5, 1980, provided preliminary clarification of TMI Action Plan requirements. The implementation of these requirements is addressed in the NRC staff testimony of Robert W. Reid concerning Board Question 5.

Seven recommendations of NUREG-0565 relate specifically to small break LOCA analysis and are listed in Table 2-1 of that document. (Items 2.2.2.a, 2.2.2.c, 2.6.2.a and 2.6.2.g concern the computer models and Items 2.2.2.b, 2.6.2.c and 2.6.2.d concern additional analyses).

NUREG-0565 Items 2.2.2.a, 2.2.2.c, 2.6.2.a, 2.6.2.g

The concerns by the B&O Task Force regarding the small break LOCA models, involve the need to confirm specific model features against applicable experimental test data. The recent tests against which present small-break LOCA models can both qualitatively and quantitatively assessed include the entire semiscale small-break test series and LOFT test L3-1 and L3-2. Other separate effects tests, (e.g., ORNL core uncovery tests) and future tests, as appropriate, should also be factored into this assessment. As discussed in the NRC response to UCS contention 8, a considerable margin exists between the calculated core

conditions for postulated small breaks at TMI-1 and the core damage limits of 10 CFR 50.46. The NRC staff believes that further refinement of the small break LOCA models is desirable in understanding the sequence of events during the accident but that the current model and calculations are in conformance with 10 CFR 50.46.

NUREG-0565 Items 2.2.2.b, 2.6.2.c, 2.6.2.d

If model deficiencies are discovered as a result of the above data comparisons, the NRC will require that the ECCS models be revised and the small-break spectrum analyses for TMI-1 be repeated (Item 2.2.2.b). The additional analyses recommended in Items 2.6.2.c and 2.6.2.d involve multiple system failures which would fall within the range of the break spectrum already analyzed. Operator action in dealing with these events will be included in Task I.c.1 of the TMI-1 Action Plan.

NUREG-0623

The staff review of analyses by the reactor vendors regarding tripping of the reactor coolant pumps is described in NUREG-0623. These analyses predict that if the coolant pumps are tripped during certain periods during a small break LOCA a greater degree of core uncovery could occur than if the pumps were tripped immediately. The staff therefore issued IE Bulletin 79-05C to the owners of B&W plants requiring an immediate manual pump trip following an indication of HPI actuation and requiring an additional operator to be in the control room to perform the action. The manual trip requirement is reflected in the TMI-1 small break LOCA procedures and the NRC concluded that TMI-1 was in conformance with the short-term requirements of IE Bulletin 79-05C.

Bulletin 79-05C also contained a long-term requirement for automatic tripping of the reactor system pumps. The automatic pump trip was recommended in NUREG-0623 so that the pumps would be tripped for LOCA events but would not be tripped for non-LOCA events. NUREG-0623 recognized an uncertainty in the thermal-hydraulic phenomenological modeling of small breaks with the pumps running. To evaluate this uncertainty Action Plan Item II.K.3.5 of NUREG-0660 was revised in May 1980 to require a continued study of the criteria for early reactor coolant pump trip. Holders of approved ECC models have been required to analyze the forthcoming LOFT test (L3-6). The capability of the industry models to corectly predict the experimental behavior of this test will have a strong input on the staff's determination on when and how the reactor coolant pumps should be tripped. The implementation of the Action Plan is discussed in the NRC testimony of R. Reid to Board Question 5.

The assumption that the operator manually trips the reactor coolant pumps immediately following a small break LOCA is the only reliance on non safety grade equipment and the only operator action assumed in the analyses of small break LOCAs described in the NRC response to UCS Contention 8. As discussed above an additional operator will be available in the TMI-1 control room to trip the reactor coolant pumps. The operators will be trained to perform this action as discussed on page C1-16 of NUREG-0680. Four operational transients in PWRs (North Anna Unit 1, Prairie Island, Arkansas Nuclear One Unit 2, and Crystal River 3) which occurred in 1979 and 1980 have indicated that operators have acted promptly in tripping the reactor coolant pumps when safety injection signals were received at those facilities. The NRC staff believes that the manual trip requirement at TMI-1 is adequate in the interim period while the need for an automatic trip is evaluated under the Action Plan.

WALTON L. JENSEN, JR.

PROFESSIONAL QUALIFICATIONS

I am a Senior Nuclear Engineer in the Reactor Systems Branch of the Nuclear Regulatory Commission. In this position I am responsible for the technical analysis and evaluation of the public health and safety aspects of reactor systems.

From June 1979 to December 1979, I was assigned to the Bulletins and Orders Task Force of the Nuclear Regulatory Commission. I participated in the preparation of NUREG-0565, "Generic Evaluation of Small Break Loss-of-Coolant Accident Behavior in Babcock & Wilcox Designed 177-FA Operating Plants."

From 1972 to 1976, I was assigned to the Containment Systems Branch of the NRC/AEC, and from 1976 to 1979, I was assigned to the Analysis Branch of the NRC. In these positions I was responsible for the development and evaluation of computer programs and techniques to calculate the reactor system and containment system response to postulated loss-of-coolant accidents.

From 1967 to 1972, I was employed by the Babcock and Wilcox Company at Lynchburg, Virginia. There I was lead engineer for the development of loss-of-coolant computer programs and the qualification of these programs by comparison with experimental data.

From 1963 to 1967, I was employed by the Atomic Energy Commission in the Division of Reactor Licensing. I assisted in the safety reviews of large power reactors, and I led the reviews of several small research reactors.

I received an M.S. degree in Nuclear Engineering at the Catholic University of America in 1968 and a B.S. degree in Nuclear Engineering at Mississippi State University in 1963.

I am a graduate of the Oak Ridge School for Reactor Technology, 1963-1964.

I am a member of the American Nuclear Society.

I am the author of three scientific papers dealing with the response of B&W reactors to Loss-of-Coolant Accidents and have authored one scientific paper dealing with containment analysis.