UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SEP 1 2 1988

MEMORANDUM FOR: Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

FROM: Eric S. Beckjord, Director
Office of Nuclear Regulatory Research

SUBJECT: RESEARCH INFORMATION LETTER NO. 157, "REPORT ON THE RESULTS OF THE CORE I TEST SERIES AT THE JAPAN ATOMIC ENERGY RESEARCH INSTITUTE SLAB CORE TEST FACILITY"

This Research Information Letter transmits the results obtained from the Core I test series carried out at the Japan Atomic Energy Research Institute (JAERI) Slab Core Test Facility (SCTF). SCTF was intended to study the reflood stage of a Large-Break Loss-of-Coolant Accident (LBLOCA) in a Pressurized Water Reactor (PWR). This work is part of the 2D/3D International Loss-of-Coolant Accident (LOCA) Research Program. A total of 28 tests were conducted, analyzed and documented in "Research Information Report on the Results of the Core I Test Series at the Japan Atomic Energy Research Institute Slab Core Test Facility," (Attachment B) prepared by MPR Associates, Inc. Thirteen of the tests were analyzed with the TRAC computer code by the Los Alamos National Laboratory and the results documented in "Research Information Report on the TRAC Analysis and Experimental Results of the Core I Test Series at the Japan Atomic Energy Research Institute Slab Core Test Facility" (Attachment C).

The SCTF is a two-dimensional core thermal-hydraulic test facility with a full-height, full-radius (8 bundles), and one-bundle-width vessel, containing approximately 2000 electrically heated rods. It is volume scaled 1:21. The primary loop is simulated with a hot leg, steam-water separator, pump simulator, intact cold leg, and broken cold leg. The slab-shaped test vessel contains a downcomer and the upper and lower plena.

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1. Regulatory Issue

SCTF Core-1 addressed issues that arose during the hearings on ECCS performance in the early 1970's which led to adoption of 10 CFR 50 Appendix K. SCTF was developed as part of the 2D/3D Program to provide multi-dimensional data on core thermal-hydraulics. Specifically SCTF Core 1 studied: reflood heat transfer; effect of flow blockage as would result from clad ballooning; and hot channel effects and resulting cross flows and recirculating flows. There was a concern that two-phase mixtures may seek preferential flow paths during the reflooding process, thereby creating the possibility that a certain region of the core may be deprived of water and thus overheated. The research to address these issues was endorsed by NRR (as well as the Commission), including a request that the RES-developed computer code TRAC be verified with respect to its ability to calculate multidimensional flows (Attachment A).

2. Research Results Observed

SCTF showed that hot channel effects caused by variations in power density are effectively mitigated by cross flows between bundles. The collapsed two-phase mixture level (i.e., liquid inventory) is essentially constant across the core radius during reflood. In the region below the quench front, liquid flows from neighboring bundles toward the high power bundle while above the quench front, a two-phase mixture flows away from the high power bundle toward the low power bundles. The shape of the power density profile does not degrade core cooling as compared to the reference test where a flat power profile was used. This finding is true for at least up to 40% power skew.

The SCTF Core I tests showed dispersed liquid throughout the core soon after the ECC liquid reaches the bottom of the core; 5-10% liquid volume fraction was observed in all core regions above the quench front within 10-20 sec. after reflood initiation. The quick disperal of liquid resulted in only a small temperature rise subsequent to the beginning of reflood.

Liquid carried over from the core and de-entrained in the upper plenum formed a pool above the upper core support plate; in the base-case test the height of the pool varied from 0.5m above Bundle 1, which represented a high power bundle in the center of reactor vessel, to 0.75m above Bundle 8, which represented a low power bundle in the periphery of the reactor vessel. The liquid level is progressively higher along a radial direction toward the hot leg probably because liquid droplets entrained in the steam are further de-entrained as they pass through the upper plenum structure on their way to the hot leg. It appears that neither the variation of 0.25m in liquid level nor the non-uniformity of structural elements resulted in any appreciable preferential flow path.

The TRAC computer code predicts two-dimensional flow behavior in the core consistent with the experimental data. TRAC predicted peak clad temperatures within 100K for those tests which were analyzed.

The SCTF-I tests showed that the effect of 60% coplanar blockage on peak clad temperatures was negligible. Even though only two out of 8 bundles had blockage, the flow bypass around the blocked bundles was not appreciable.

An open vent valve (simulating B&W plants) between the upper plenum and the top of the downcomer produced slightly better core cooling than the closed vent valve, as expected.

3. Conclusions and Regulatory Implications

Based on the SCTF I tests, we conclude the following:

A. The core cooling may be represented with a one dimensional model along the axial direction. The core cooling in the radial direction may be adequately represented with average quantities as long as the radial variations in power stay within a 40% band about the average power.

B. The sixty percent coplanar flow blockage in the core has a negligible effect on core cooling.

C. Some of the features of 10 CFR 50 Appendix K were confirmed as being conservative. These features are low containment pressure, locked pump impeller, no steam flow in the intact loops during the accumulator injection period, and steam cooling only for a flooding rate less than 1 inch/sec, and low ECC flow rate resulting from one pump failure. However, the effects of these conservatisms on peak clad temperatures were not significant except for the steam-cooling-only feature.

4. Restriction on Application

The SCTF is a low pressure test facility which operated at less than 6 bar. In addition, there are other facility limitations such as an unrealistic square downcomer, a narrow, oval-shaped hot leg, and the excessive influence of vessel wall. These factors must be considered before the SCTF results are applied to a PWR. Usually, the TRAC or a similar computer code must be utilized for relating SCTF results to the full scale PWR. However, as far as the above-stated conclusions are concerned, they are considered applicable to the full scale PWR.

5. Unresolved Questions

There are no unresolved questions with respect to two regulatory issues discussed above; namely (1) the possibility of developing preferential flow paths in the core during the reflood phase of a LBLOCA and (2) the effect of 60% coplanar flow blockage in the core. The answers have been provided as discussed above.

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Enclosures:
A. NRR Comments on Extension of 2D/3D Agreement beyond April 1985.
B. Research Information Report on the Results of the Core I Test Series at the Japan Atomic Energy Research Institute Slab Core Test Facility.
C. Research Information Report on the TRAC Analysis and Experimental Results of the Core I Test Series at the Japan Atomic Energy Research Institute Slab Core Test Facility.