March 19, 1996

Mr. James M. Taylor
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Mr. Taylor

SUBJECT: NRC STAFF PROGRAM ON THE ADEQUACY ASSESSMENT OF THE RELAP5/MOD3 CODE FOR SIMULATION OF AP600 PASSIVE PLANT BEHAVIOR

During the 429th meeting of the Advisory Committee on Reactor Safeguards, March 7-9, 1996, we reviewed the program being conducted by the Office of Nuclear Regulatory Research (RES) to assess the adequacy of the RELAP5/MOD3 code for simulating the behavior of the Westinghouse AP600 passive plant design. During this review, we had the benefit of discussions with representatives and consultants of the NRC staff and the Idaho National Engineering Laboratory (INEL). Our Subcommittee on Thermal Hydraulic Phenomena held a meeting on this matter on February 22-23, 1996. We also had the benefit of the referenced documents.

We have been asked to comment on the approach and methodology for demonstrating the adequacy of the RELAP5/MOD3 code to calculate AP600 passive plant behavior in support of the design certification review. We believe that the overall approach and methodology being employed by RES for this assessment is acceptable. Most of the necessary elements are in place. A substantial amount of work remains, however, and we believe that the schedule for successful completion cannot be met.

Our comments and recommendations relative to this review, primarily based on oral presentations, are:

Since we last reviewed this program in 1994, significant improvements have been made. The most significant has been the increased emphasis on the code improvement program. Other changes that have led to excellent results include the involvement of outside technical expertise, via the Thermal Hydraulic Expert Consultants group and the direct involvement of RES technical personnel in the research activities. Particularly noteworthy accomplishments include the analysis of water hammer, the treatment of flow oscillations observed in the tests during injection from the In-containment Refueling-Water Storage Tank and the evaluation and explanation of strong thermal stratification in the ROSA cold leg.

RES should perform a more robust and complete top-down system scaling analysis for ROSA, SPES, and OSU. An entire transient should be evaluated to quantify the effects of various distortions

in the three facilities and to demonstrate that the experimental database is sufficient to validate the code. Any additional distortions or anomalies identified should be added to the list of distortions compiled by RES in late-1994, and that remain to be addressed. The scaling effort should be integrated with the Phenomena Identification and Ranking Table.

The thermal stratification that was seen in ROSA tests for a one-inch cold-leg break was initially identified as a potentially important safety issue for the AP600. It has now been shown to be just a manifestation of scale distortion in the ROSA facility. This demonstrates the need to identify and explain anomalous behavior.

The thermal stratification in the Core Makeup Tank (CMT) observed in the tests needs to be studied. Its effects on core inventory have to be understood because neither RELAP5/MOD3 nor the Westinghouse computer codes can, at present, reliably predict thermal stratification.

The screening study for water hammer in the AP600 design addressed an important safety issue. The study allows an analysis of the potential for such events and provides a method for estimating the resulting loads in susceptible areas. We recommend that this study be published soon as a separate report.

The documentation provided for our review did not, by itself, furnish an adequate basis upon which we could logically endorse the process. The documentation provided to the Thermal Hydraulic Phenomena Subcommittee in advance of the February 22-23, 1996 meeting was inconsistent and contained results declared incorrect by RES during the meeting. Furthermore, the RELAP5/MOD3 Code Manual published in August 1995 was not provided to us in time to support our review.

RELAP5 is still undergoing significant and rapid modifications. A calculation has not yet been performed with a version of the code that contains all the planned changes. Numerous calculations will need to be performed to mature the code and validate it using data obtained from various separate effects and integral facilities tests.

Overall, the approach and methodology for qualifying RELAP5/MOD3 for AP600 simulation appear to be adequate. However, two possible "show stoppers" remain: 1) simulation of the CMT thermal stratification and 2) simulation of long-term cooling, which is still an issue. Serious consideration should be given to addressing these obstacles.

Dr. George Apostolakis did not participate in the Committee's deliberations of this matter.

Sincerely,

/s/

T. S. Kress
Chairman

References:

1. Memorandum dated January 22, 1996 from M. W. Hodges, Office of Nuclear Regulatory Research, NRC, to J. Larkins, Advisory Committee on Reactor Safeguards, NRC, transmitting:

   - Volume 2 of 10 volumes of adequacy demonstration reports, "Adequacy Assessment Overview"

   - Idaho National Engineering Laboratory draft report prepared for U.S. Nuclear Regulatory Commission, "Adequacy Evaluation of RELAP5/MOD3 for Simulating AP600 Small Break Loss-of-Coolant Accidents, Volume 2: Horizontal Integrated Analysis of the AP600 1-Inch Diameter Cold Leg Break," November 1995, with Appendices A-K (Proprietary)

2. Idaho National Engineering Laboratory, draft report prepared for U.S. Nuclear Regulatory Commission, "Top-Down Scaling Analysis Methodology for AP600 Integral Tests," January 1996

3. Letter report dated April 12, 1995, to James M. Taylor, Executive Director for Operations, NRC, from T. S. Kress, Chairman, Advisory Committee on Reactor Safeguards, Subject: NRC Test and Analysis Program in Support of AP600 Advanced Light Water Passive Plant Design Review

4. Letter dated May 8, 1995, from James M. Taylor, Executive Director for Operations, NRC, to T. S. Kress, Chairman, Advisory Committee on Reactor Safeguards, Subject: Staff Response to ACRS Letter Dated April 12, 1995, on NRC Test and Analysis Program in Support of AP600 Advanced Light Water Passive Plant Design Reviews

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