March 19, 1997

Mr. L. Joseph Callan
Executive Director for Operations
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555-0001

Dear Mr. Callan:

SUBJECT: NRC OFFICE OF NUCLEAR REGULATORY RESEARCH PROGRAM TO DEMONSTRATE THE ADEQUACY OF THE RELAP5/MOD3 CODE TO ANALYZE AP600 PASSIVE PLANT BEHAVIOR

During the 439th meeting of the Advisory Committee on Reactor Safeguards, March 6-8, 1997, we completed our review of the NRC Office of Nuclear Regulatory Research (RES) program to demonstrate the adequacy of the RELAP5/MOD3 code for analyzing the behavior of the Westinghouse AP600 passive plant design. Our Subcommittee on Thermal-Hydraulic Phenomena met to consider this matter on February 22-23, 1996 and February 12-14, 1997. We previously commented on our initial review of this issue in a March 19, 1996 letter to the Executive Director for Operations. During this review, we had the benefit of discussions with representatives of the NRC staff, and we also had the benefit of the documents referenced.

Scaling

The staff sponsored two scaling analyses to show that data obtained from several integral and separate effects test facilities are applicable to AP600 and can be used to validate RELAP5/MOD3.2.1.2 for assessing the ability of the AP600 design to address design basis accidents (DBAs). Both sets of analyses demonstrated that the data available are appropriate for validating the code. We agree with the conclusions of these analyses and compliment the staff and its contractors on a job well done. A final summary report consolidating the results of these two analyses should be written to document the conclusion that the data are sufficiently complete and also to note how the tests were used for code validation.

Code Adequacy

The code documentation made available to us was out of date. As a result, our conclusions about code adequacy are based partly on the additional information presented at the meetings. Based on our review of this information, we believe that RELAP5/MOD3.2.1.2 is adequate for evaluating Westinghouse AP600 DBAs. The RELAP5/MOD3.2.1.2 calculations for particular DBAs showed that sufficient margin exists in the AP600 design. The tests and the associated code predictions indicate that the core does not uncover under any of the tested DBA conditions. Documentation reflecting what was presented to us should be completed in a timely manner. We also believe that there is a need to develop more quantitative criteria than those used by RES for assessing the adequacy of a thermal-hydraulic code.

During our review, a number of deficiencies in the RELAP5/MOD3.2.1.2 code were identified but were determined not to significantly affect the code's capability to evaluate the AP600 emergency core cooling system behavior. Nevertheless, these deficiencies indicate that the code falls short of being a true best-estimate code. (These deficiencies are also generally found in the TRAC code.) The deficiencies identified include the following:

1. The critical flow model is a "work-around" that is appropriate only for this specific application.

2. The treatment of subcooled nucleate boiling in the core is poor.

3. The code does not model horizontal stratified flow very well.

4. The drift-flux correlations used in the code have known deficiencies at low pressure.

5. The code inadequately treats condensation and thermal stratification.

6. Treatment of the automatic depressurization system valves in both the code and in the experimental studies is overly simplistic.

There is a need to develop sound models in these areas for the future consolidated systems code, and we recommend that a modest effort be continued to also improve these areas in the RELAP5 code.

Inevitably, there will be uses of RELAP5/MOD3 other than demonstrating adequacy of the AP600 design. These uses could, for example, include studies ranging from loss of coolant accidents and transients, to simulator fidelity evaluation. For some of these, it is sufficient to have a conservative analysis, whereas for others the results will need to be as realistic as possible. For any such additional uses of the code, the effects of the identified deficiencies need to be evaluated.

Modeling a nuclear power plant is a complex task that requires a mix of first principles combined with a great deal of physical insight and engineering judgment. As a result, personnel who are knowledgeable about the thermal-hydraulic behavior of single and two-phase flow as well as the use of computers must be available if sound judgments are to be made.

The thermal-hydraulics team assembled by the NRC for this exercise demonstrated a great deal of understanding relative to both the AP600 test results and the accident behavior of the nominal plant design. We believe that this work is excellent preparation for the current effort of the NRC staff to develop a new "consolidated" thermal-hydraulics code and augurs well for its success.

Dr. George Apostolakis did not participate in the committee's deliberations regarding this matter.

Sincerely,

/s/

R. L. Seale
Chairman

References:

1. U. S. Nuclear Regulatory Commission Report, NUREG/CR-5535, "RELAP5/MOD3 Code Manual," Volumes 1-7, August 1995.
2. Letter dated March 19, 1996, from T. S. Kress, Chairman, ACRS, to J. M. Taylor, Executive Director for Operations, NRC, Subject: NRC Staff Program on the Adequacy Assessment of the RELAP5/MOD3 Code for Simulation of AP600 Passive Plant Behavior.
3. Memorandum dated December 27, 1996, from M. W. Hodges, Office of Nuclear Regulatory Research, NRC, to J. T. Larkins, Executive Director, ACRS, Subject: Adequacy Assessment of the RELAP5 MOD3 Code to Analyze AP600 Passive Plant Behavior Following Postulated Small Break Loss-of-Coolant Accidents, transmitting the following reports:
    a. Idaho National Engineering Laboratory, Final Draft Report, INEL-96/0400, "Adequacy Evaluation of RELAP5/MOD3, Version 3.2.1.2 for Simulating AP600 Small Break Loss-of-Coolant Accidents," December 1996, with Appendix (Proprietary Information).
    b. Idaho National Engineering Laboratory Report, INEL-96/0395, "Evaluation and Assessment of RELAP5/MOD3 Version 3.2.1.2 for Simulating the Long-Term Phase of Small Break Loss-of-Coolant Accidents in the AP600," October 1996 (Proprietary Information).
    c. Idaho National Engineering Laboratory, Final Draft Report, INEL-96/0440, "Applicability of Selected RELAP5/MOD3 Models and Correlations to AP600 Small Break Loss-of-Coolant Accidents," November 1996 (Proprietary Information).
    d. Idaho National Engineering Laboratory, Final Draft Report, INEL-96/0117, "Top-Down Scaling Analyses of AP600 Integral Test Data in Support of RELAP5 Adequacy Assessment," December 1996 (Proprietary Information).
    e. Idaho National Engineering Laboratory Report, INEL-94/0061, "Phenomena Identification and Ranking Tables for Westinghouse AP600 Small Break Loss-of-Coolant Accident, Main Steam Line Break, and Steam Generator Tube Rupture Scenarios?" Revision 2, November 1996.
    f. U. S. Nuclear Regulatory Commission, Draft Report, "Phenomenology Observed in the AP600 Integral Systems Test Programs Conducted in the ROSA-AP600, APEX, and SPES-2 Facilities, It D. Bessette, Office of Nuclear

Regulatory Research, NRC; M. DiMarzo, University of Maryland; P. Griffith, Massachusetts Institute of Technology, December 1996 (Proprietary Information).

g. U. S. Nuclear Regulatory Commission, Draft Report, "Screening Reactor Steam/Water Piping Systems for Water Hammer," P. Griffith, Consultant, November 1996.

4. Idaho National Engineering Laboratory, Draft Report, INEL-96/0040, "Top-Down Scaling Analysis Methodology for AP600 Integral Tests," S. Banerjee, et al., January 1997 (Proprietary Information).

→