ACRSR-1663

October 21, 1996

The Honorable Shirley Ann Jackson
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Dear Chairman Jackson:

SUBJECT: THERMAL-HYDRAULICS RESEARCH PLAN

During the 435th meeting of the Advisory Committee on Reactor Safeguards, October 9-12, 1996, we reviewed the scope and approach of the Thermal-Hydraulics Research Plan of the Office of Nuclear Regulatory Research (RES). Our Subcommittee on Thermal-Hydraulic Phenomena met on September 18-19, 1996, to review this matter. During this review, we had the benefit of discussions with representatives of the NRC staff. We also had the benefit of the documents referenced.

The overall plan developed by RES to consolidate existing computational tools into a single computer code is timely and should be implemented. We agree with its objectives of standardized programming, better physics, flexibility (modularity), computational efficiency, a graphical user interface, and thorough documentation. The RES plan to review the past 22 years of experience with codes like TRAC and RELAP, as well as the successful Code Scaling Applicability and Uncertainty evaluation methodology, should help avoid some of the problems of the past. We recommend that this review also consider the French code, CATHARE, and its uncertainty evaluation methodology.

The RES staff expects to identify the key physical processes that the new code must model. Also, RES plans to determine whether TRAC-P has an architecture that will allow it to provide flexibility with respect to insertion of new models or modules and whether it has the capability to interface with other codes like CONTAIN and SCDAP. We concur in these plans and emphasize that highest priority should be given to the development of sufficient flexibility to facilitate modifications in response to future modeling challenges.

The NRC Office of Nuclear Reactor Regulation and the Office for Analysis and Evaluation of Operational Data are primary users of thermal-hydraulic codes. They should be a part of this process from the beginning. Consequently, we recommend that a code users group be instituted early in the development program.

We concur in the RES plan to incorporate the integral effects test programs at Oregon State University, the University of Maryland, and Purdue University into the overall verification and validation program. The cooperative agreement with the French authorities to obtain analytical and experimental data developed at the Grenoble facility should also prove to be valuable for validating the code. The present RES relationship with the above three universities and the French authorities should significantly enrich the proposed thermal-hydraulics research Plan. These, along with other cooperative agreements, should be pursued, independent of the final direction of the RES Plan.

We commend the staff for the development of this Plan which holds much promise to revitalize the NRC Thermal-Hydraulics Research Program.

Additional comments by ACRS Member Ivan Catton are presented below.

Sincerely,

/s/

T. S. Kress
Chairman

Additional Comments by Ivan Catton, ACRS Member

I agree with the views of my colleagues expressed above but would like to emphasize the need for careful planning at the outset of the RES Thermal-Hydraulics Research Plan. The research program that led to the present suite of thermal-hydraulic codes was initiated in 1974 to address the large-break loss of coolant accident. The mission was well defined and the agency met its objectives.

At the outset, it was thought that a properly designed thermal-hydraulic code would be able to model all related problems. Over the years, however, experience has shown that the codes did not meet this objective; i.e., whenever we needed solutions to a new problem that was a little different, the codes were inadequate, because they could not be readily modified to accommodate the special circumstances demanded by the new problem.

The inability of the codes to address numerous new problems emphasizes the need for a different approach. There is no single code that can model all the different physical phenomena that occur in a nuclear power plant. A broader approach is needed where different modeling schemes can be tied together to successfully address the problem at hand. Further, a skilled code user, who is also knowledgeable in the field of thermal-hydraulics, is needed to decide what is important and how to implement it in a code. A code, no matter how good, will never substitute for a capable thermal-hydraulic analyst.

Some of these problems will be heavily dependent on the use of what is commonly known as computational fluid dynamics (CFD), some on the use of the kind of modeling found in today's codes, and some will require an empirical approach. There will be some problems that may even require the use of stand-alone CFD codes. Further, the development of a single code for all users may not be a realistic goal. A skilled user needs a different level of computational power than does a less-skilled user. Ensuring adequate flexibility in a single code to accommodate the needs of both computational power and user skills will require a great deal of thoughtful planning; this planning should take place at the beginning of the development of the RES Thermal-Hydraulics Research Plan.

References:

1. Memorandum dated September 6, 1996, to the Commission from James M. Taylor, NRC Executive Director for Operations, Subject: Thermal-Hydraulic Five-Year Research Plan, (Predecisional - For Internal Use Only)
2. U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Analytical Support Group, Technical Analysis Reports, SASG-94-01 - SASG-94-05; SASG-95-01 - SASG-95-07; SASG-96-01 - SASG-96-07 (Proprietary Information)
3. ACRS report dated June 15, 1989, from David A. Ward, ACRS Chairman, to Lando W. Zech, Jr., NRC Chairman, Subject: NRC Thermal-Hydraulic Research Program
4. ACRS report dated June 7, 1988, from David A. Ward, ACRS Chairman, to Lando W. Zech, Jr., NRC Chairman, Subject: NRC Research Related to Heat Transfer and Fluid Transport in Nuclear Power Plants