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NOTE TO EDITORS:

The Nuclear Regulatory Commission's Executive Director for Operations, James M. Taylor, has received the attached report from the NRC's Advisory Committee on Reactor Safeguards. The report provides comments on NRC test and analysis programs in support of design certification reviews for the Westinghouse Electric AP600 and GE Nuclear Energy simplified boiling water reactor advanced light water reactors.

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Attachment:
As stated

November 10, 1994

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

Dear Mr. Taylor:

SUBJECT: NRC TEST AND ANALYSIS PROGRAMS IN SUPPORT OF AP600 AND SBWR ADVANCED LIGHT WATER REACTOR PASSIVE PLANT DESIGN CERTIFICATION REVIEWS

During the 414th and 415th meetings of the Advisory Committee on Reactor Safeguards, October 6-7 and November 3-4, 1994, we discussed the confirmatory test and analysis programs being conducted by the Office of Nuclear Regulatory Research (RES) in support of the design certification reviews for the Westinghouse AP600 and GE Nuclear Energy (GENE) Simplified Boiling Water Reactor (SBWR) advanced light water reactors. During these meetings, we had the benefit of discussions with representatives of RES. Our Subcommittee on Thermal Hydraulic Phenomena held a meeting on August 25-26, 1994, to discuss this matter. We also had the benefit of the documents referenced.

In the absence of a full-scale test facility, an understanding of the thermal hydraulic behavior of a passive plant design will depend on the use of computer codes. The NRC staff has decided to modify RELAP5/MOD3 for its confirmatory thermal hydraulic analysis of the AP600 and SBWR designs. The important phenomena the code must simulate should be delineated in the Phenomena Identification and Ranking Table (PIRT), thus allowing one to formulate integral and separate effects experiments that will yield appropriate data for code validation. Code validation should be an integrated process involving code development, experimentation, and an understanding of the physics of two-phase flow and heat transfer.

The major objective of the thermal hydraulic code development effort should be to produce a code capable of predicting the behavior of a full-scale nuclear power plant with acceptable uncertainties. For existing nuclear plant designs, we have had the benefit of many integral and separate effects experiments at a wide variety of scales to help arrive at an estimate of the uncertainties in the code predictions. We are now dealing with two passive plant designs which evidence more complex thermal hydraulic system dynamics, and for which there is a paucity of relevant experimental data. There are several causes for this more complex dynamic behavior: (1) steam condensation at low pressure, (2) use of gravity-driven coolant injection, and (3) the existence of many components and complex hydraulic paths that give the system many degrees of freedom. Understanding this dynamic behavior requires evaluation of scale distortion effects and dynamic characteristics in the various test facilities. In this regard, two questions should be addressed and resolved: (1) is the evolution of a particular transient influenced by configurational and/or scale distortions, and (2) do configurational and/or scale distortions in the various test facilities preclude simulation of some important dynamic effects while introducing other dynamic effects that may not be important in a full-scale plant design? To address these questions, a top-down scaling analysis must be performed.

The NRC staff has test and analysis programs under way to address issues arising during its evaluation of the AP600 and the SBWR designs. The AP600 evaluation will be supported by testing at the Japan Atomic Energy Research Institute ROSA-V facility and the use of RELAP5/MOD3. The SBWR evaluation will be supported by testing at the Purdue University PUMA facility and the use of RELAP5/MOD3. We believe that the use of RELAP5/MOD3 for both AP600 and SBWR simulations will lead to the development of a more robust computational tool. Both programs are discussed below and some comments about the technical direction of these programs are provided.

AP600 Program

The PIRT in support of the AP600 analysis has not yet been completed. There is no indication that a PIRT was utilized for allocating resources, for assigning test objectives, or for developing the test matrices. It is necessary to complete the PIRT and confirm it on the basis of relevant scaling groups. To ensure that RELAP5/MOD3 can simulate the high ranking phenomena, specific tests in the test matrix should be associated with the high ranking phenomena in the PIRT. By doing this, all important phenomena will be addressed.

The PIRT and a proper scaling analysis for the AP600 would cover all test facilities for AP600. Unfortunately, the scaling efforts conducted for the OSU, SPES, and ROSA-V test facilities were not coordinated. The global scaling of the AP600 design, including consideration of the dynamic interactions between the major system components (pressure vessel, core makeup tank, pressurizer, steam generators, passive residual heat removal system, and accumulators), was omitted. Depressurization is not scaled, even though the methods for doing so are known. The scaling analysis for OSU, while still incomplete, could serve as a model for ROSA and SPES.

Direct counterpart tests in ROSA, OSU, and SPES are not possible. This makes it difficult to extrapolate the observed thermal hydraulic behavior to full scale. A well-planned effort to integrate experiments with code improvement and assessment is needed to quantify uncertainties. At present, RELAP5/MOD3 predicts strong oscillations both when they are observed in tests and when they are not. Consequently, the calculated behavior can neither be attributed conclusively to numerical nor physical effects. The mechanisms by which the various observed modes of oscillation are initiated and maintained need to be understood so that their potential influence on the thermal hydraulic behavior of the AP600 can be evaluated. The judicious selection of test conditions for the facilities, together with the conduct of a careful data analysis and scaling, should provide a satisfactory solution.

The demonstrated propensity for condensation oscillation events in the AP600 points to a need to identify both the likelihood and damage potential of water hammer events. Furthermore, the influence of thermal stratification on the thermal hydraulic behavior of the AP600 also remains to be evaluated.

SBWR Program

The objective of the PUMA test program is to obtain data for assessing computer code simulation of important SBWR-specific phenomena. The focus of this test program is on the operability of the passive cooling systems and their interactions with the reactor vessel.

Again, a PIRT has not been completed. The PIRT effort should be brought to a close so that a proper evaluation of PUMA and the GENE test facilities (GIST, GIRAFFE, and PANDA) can be made.

Scaling of phenomena identified in the Purdue University preliminary PIRT has been a major part of the PUMA test program. At present, the scaling effort has primarily focused on the details of local phenomena whereas global scaling appears to be incomplete. To preclude atypicalities in the interactions of the various systems and to help determine an appropriate set of initial and operating conditions for the PUMA system, the scaling of the global dynamic component interactions (among the reactor vessel, drywell, wetwell, PCCS, ICS, and GDCS) should be completed before the facility design is frozen.

We are pleased to see that one of the PUMA program principal investigators is a code developer. Input from a code developer on the selection of instrument type, number, and location will yield a much more useful set of data for code assessment.

The PUMA facility will allow testing that both overlaps and extends the accident period covered by the GENE test facilities (GIST, GIRAFFE, and PANDA), while allowing the simulation of a broad spectrum of postulated accidents. This should be helpful in confirming the validity of the results obtained at the GENE facilities.

The following comments are specific to the PUMA program:

- The current plan is to measure the heat transfer characteristics and infer the noncondensible gas concentration. We would like to point out that knowledge of the noncondensible gas distribution is fundamental and necessary if one is to avoid compensating errors in the computational process. We recommend that the noncondensible gas concentration be measured directly at several locations.

- The test matrix does not include a long-duration test. We believe it should because the SBWR containment performance requirement is 72 hours, which scales to 144 hours of PUMA test time.

- Since the interface temperature of the suppression pool is directly coupled to the containment pressure, an evaluation of thermal stratification in the pool is needed.

- Some tests should be conducted with initial nitrogen concentrations in the drywell to evaluate the impact of steam line breaks outside containment.

- The planning of the PUMA experiments should include consideration of phenomena arising as a consequence of failures of active mitigating systems.

- Data analysis and evaluation are not part of the contract with Purdue University. This is unfortunate because in this case the principal investigators at Purdue University are highly qualified for such a task. Further, those conducting the testing can bring valuable insights to the process. We recommend that the contract with Purdue University be modified to include a data analysis and evaluation task.

Technical Oversight

The RES staff now plans technical oversight of thermal hydraulic research for the AP600 and the SBWR through the Advanced Light Water Reactor Thermal Hydraulic Research Integration Group (ATRIG). This unwieldy ATRIG is not the technical oversight recommended by the ACRS in the past and subsequently approved by the Commission. Lessons learned from the CSAU program should be remembered. A small (5 or 6 members) cohesive group with well-qualified leadership is needed to integrate the technical issues of scaling, data collection, data analysis, and code development.

Sincerely,

T. S. Kress
Chairman

<u>References</u>:

1. "Summary of the LSTF Characterization Tests Performed in Conjunction with the ROSA/AP600 Experiments," R. A. Shaw, et al., Draft report dated August 1, 1994, transmitted by memorandum dated August 5, 1994, from G. S. Rhee, Office of Nuclear Regulatory Research
2. U. S. Nuclear Regulatory Commission, Draft NUREG/CR, PU-NE 94/1, Subject: Scientific Design of Purdue University Multi-dimensional Integral Test Assembly (PUMA) for GE SBWR, July 1994, transmitted by memorandum dated August 4, 1994, from J. T. Han, Office of Nuclear Regulatory Research
3. Memorandum dated August 8, 1994, from M. Ishii, Purdue University, to J. Han, U. S. Nuclear Regulatory Commission, transmitting replacement pages for the report, "Preliminary Scientific Design of Purdue University Multi-dimensional Integral Test Assembly (PUMA) for GE SBWR"
4. U. S. Nuclear Regulatory Commission, NUREG/CR-6066, EGG-2705, "Scaling and Design of LSTF Modifications for AP600 Testing," T. J. Boucher, et al., August 1994
5. SECY-94-138, memorandum dated May 20, 1994, from James M. Taylor, Executive Director for Operations, NRC, for the Commissioners, Subject: Confirmatory High Pressure Integral System Testing of the Westinghouse AP600 Safety Systems
6. "Quick Look Report for ROSA/AP600 Experiment AP-CL-03," R. A. Shaw, et al., undated rough draft, transmitted by memorandum dated August 5, 1994, from G. S. Rhee, Office of Nuclear Regulatory Research
7. Advisory Committee on Reactor Safeguards Report, dated November 18, 1993, from J. Ernest Wilkins, Jr., ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: NRC Confirmatory Test Program in Support of the AP600 Design Certification