

March 17, 1997

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

Dear Mr. Callan:

SUBJECT: PUBLICATION OF PROPOSED JOURNAL ARTICLE CONTAINING ROSA
TEST DATA

In a December 13, 1996 letter, Mr. James M. Taylor (then Executive Director for Operations) requested that the ACRS provide an independent review of the technical merits of a Westinghouse Electric Corporation concern that the data from some of the ROSA-V tests included in a proposed journal article jointly authored by NRC and Japan Atomic Energy Research Institute personnel when combined with other publicly available information could permit a competitor to deduce proprietary design details through "reverse engineering."

During our 439th meeting, March 6-8, 1997, we reviewed the merits of this concern. Our Subcommittee on Thermal-Hydraulic Phenomena also reviewed this issue during a meeting on February 19, 1997. During this review, we had the benefit of discussions with representatives of the NRC staff and Westinghouse Electric Corporation. We also had the benefit of the documents referenced.

During the February 19, 1997 Subcommittee meeting, representatives of Westinghouse stated that their concern was somewhat broader than that posed to the Committee by Mr. Taylor. Their broader concern was that a third party could use information as presented in the proposed article along with other available information to provide credibility to a competing design and thereby undermine the competitive international marketing position of Westinghouse.

We recognize that there are elements of this issue that will not be captured by a strictly technical review. These could include, for example, NRC needs for (1) sufficient public disclosure of its safety case, (2) independence from the vendors, and (3) uninhibited future publication of data from other test facilities. Nevertheless, in our review, we have chosen to focus on two technical questions associated with the broader concern:

- (1) Is there sufficient information available in the proposed article (in its current form) that could be used along with other information in the public domain by a knowledgeable third party to develop a technically defensible thermal-hydraulic computer mockup of the ROSA-V facility?

- (2) If so, can the data in the form currently proposed to be published in the article be used along with a computer model to provide credibility to a competing design for which the performance of the passive systems is demonstrated via the computer model?

It is our opinion that a technically defensible computer model of the ROSA-V facility could be developed by a third party from the information in the proposed article when combined with other available information in the public domain, and that such a model could be an important factor in providing credibility to a competing passive design. A claim could be made that the design is based on an NRC-approved design. In our opinion, there is significant merit to the concerns of Westinghouse and these concerns ought to be given appropriate consideration in any decision regarding the form of data presentation in the proposed article.

Sincerely,

/s/

R. L. Seale
Chairman

References

1. Letter dated December 13, 1996, from James M. Taylor, Executive Director for Operations, NRC, to Thomas S. Kress, Chairman, ACRS, Subject: ACRS Review of ROSA and APEX Data Prior to Publication.
2. Proposed journal article, "Implications of the Rig of Safety Assessment/AP600 High- and Intermediate-Pressure Test Results," by Louis M. Shotkin, Nuclear Regulatory Commission, and Yutaka Kukita, Japan Atomic Energy Research Institute (Review for Public Release Pending).
3. Letter dated February 4, 1997, from Brian A. McIntyre, Westinghouse Electric Corporation, to Robert Seale, Chairman, ACRS, Subject: ACRS Review of the Proposed Publication of a Journal Article Concerning AP600 Test Results.
4. Proposed journal article, "Core Markup Tank Behavior Observed During the ROSA-AP600 Experiments," by T. Yotomot, Japan Atomic Energy Research Institute, et al. (Review for Public Release Pending).
5. Proposed paper, "ROSA-AP600 Experiment Simulating Steam Generator Tube Rupture Transient," by H. Nakamura and Y. Kukita, Japan Atomic Energy Research Institute, for the 2nd International Topical Meeting on Advance Reactors Safety, Orlando, Florida, June 1-4, 1997.
6. Proposed paper, "Analysis of Wall Heat Capacity Effect on Core Makeup Tank Drain-Down Behavior in ROSA/AP600 Experiment," by Masaya Kondo, Japan Atomic Energy Research Institute, et al., for the 2nd International Topical Meeting on Advance Reactors Safety, Orlando, Florida, June 1-4, 1997.
7. American Nuclear Society proceedings paper, "Passive Residual Heat Removal System Heat Exchanger Characterization During

- Simulated Station Blackout," by Owen Stevens and Jose N. Reyes, Jr., Oregon State University, from the 1996 National Heat Transfer Conference, Houston, Texas, August 3-6, 1996.
8. American Nuclear Society proceedings paper, "Evaluation of the APEX Break Flow Measurement System During Subcooled Depressurization," by David A. Pimentel and Jose N. Reyes, Jr., Oregon State University, from the 1996 National Heat Transfer Conference, Houston, Texas, August 3-6, 1996.
 9. International Atomic Energy Agency proceedings paper, "SPES-2. AP600 Integral Systems Test Results," by L. E. Conway and R. Hundal, Westinghouse Electric Corporation, Italy, May 1995.
 10. Proceedings paper, "Comparison of the SPES-2 Pre-Test Predictions and AP600 Plant Calculations Using RELAP5/MOD3," by A. Alemberti, C. Frepoli, and G. Graziosi, ANSALDO Nuclear Division (Copies available from the NRC Office of Nuclear Reactor Regulation).
 11. Proceedings paper, "SPES-2 Cold Leg Break Experiments: Scaling Approach for Decay Power, Heat Losses Compensation and Metal Heat Release," by A. Alemberti, C. Frepoli, and G. Graziosi, ANSALDO Nuclear Division (Copies available from the NRC Office of Nuclear Reactor Regulation).
 12. Proceedings paper, "SPES-2, The Full-Height, Full-Pressure, Integral System AP600 Test Facility," by M. Bacchiani, SIET S.P.A., et al., Twenty-Second Water Reactor Safety Information Meeting, Bethesda, Maryland, October 24-26, 1994.
 13. Proceedings paper, "SPES-2 RELAP5/MOD3 Noding and 1 Inch Cold Leg Break Test S00401," by A. Alemberti, C. Frepoli, and G. Graziosi, ANSALDO Nuclear Division, Twenty-Second Water Reactor Safety Information Meeting, Bethesda, Maryland, October 24-26, 1994.
 14. Abstract of thesis, "Characterization of the Advanced Plant Experiment (APEX) Passive Residual Heat Removal System Heat Exchanger," by Owen Stevens, Oregon State University, presented June 7, 1996.
 15. Abstract of thesis, "Two-Phase Fluid Break Flow Measurements and Scaling in the Advanced Plant Experiment (APEX)," by David Alan Pimentel, Oregon State University, presented on May 9, 1996.

→