UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

March 24, 1999

Dr. William D. Travers
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Dr. Travers:

SUBJECT: APPLICATION OF WESTINGHOUSE BEST-ESTIMATE LOSS-OF-COOLANT ACCIDENT ANALYSIS METHODOLOGY TO UPPER PLENUM INJECTION PLANTS

During the 460th meeting of the Advisory Committee on Reactor Safeguards, March 10-13, 1999, we reviewed the Westinghouse Electric Company's application of its best-estimate loss-of-coolant accident (LOCA) analysis methodology to plants with Upper Plenum Injection (UPI). Our Subcommittee on Thermal-Hydraulic Phenomena reviewed this matter on December 16, 1998, and February 23, 1999. We also had the benefit of the documents referenced.

The best-estimate LOCA analysis methodology, which utilizes the WCOBRA/TRAC code, has been approved for use in Westinghouse three- and four-loop pressurized water reactors (PWRs). Westinghouse is requesting NRC staff approval to apply this methodology to analysis of large-break (LB) LOCAs in its two-loop plants equipped with UPI of low-pressure emergency coolant. The staff intends to approve the request. This decision is based on the results of a contractor review and the staff's assessment of the methodology, which indicate that Westinghouse has followed the steps described in the Code Scaling, Applicability, and Uncertainty (CSAU) evaluation methodology, met the intent of Regulatory Guide 1.157, and satisfied the emergency core cooling system (ECCS) Rule criteria (10 CFR 50.46 and Appendix K). We note that Regulatory Guide 1.157 allows the staff considerable latitude in deciding on the acceptability and appropriateness of the supporting evidence and analyses.

Conclusions and Recommendations

1. We agree that the results of UPI tests and analyses, as presented by Westinghouse and the Office of Nuclear Regulatory Research, show that UPI plants as currently configured and operated are likely to keep the core cooled following a LBLOCA.

2. WCOBRA/TRAC UPI code predictions of peak cladding temperatures are either conservative or appear insensitive to details in the modeling. We have three concerns:

    - It is not clear that the code can be characterized fairly as "best-estimate" or "realistic" when applied to UPI plants.

- The CSAU evaluation methodology has been carried out in a way that marginally meets the intent of the process.

- Experimental data and sensitivity studies cover a limited range. In the Safety Evaluation Report (SER) the staff should caution that applications of the code be limited to conditions representative of those tested, such as the rates of steam flow in the Upper Plenum Test Facility (UPTF); otherwise, more extensive sensitivity studies and uncertainty calculations should be considered.

3. The NRC staff needs to develop a more proactive, comprehensive, and structured process to support the review of thermal-hydraulic codes.

Discussion

Evidence for the effectiveness of UPI is based on one larger-than-full-scale UPTF test, in which ECCS water penetrated to the simulated lower plenum for conditions representative of a LB LOCA, and two Cylindrical Core Test Facility scaled tests in which a simulated core was cooled at least as well as in corresponding cold-leg injection tests. Westinghouse was able to model these tests reasonably well with its code. Westinghouse also validated its modeling of the countercurrent flow limit (CCFL) against separate-effects tests of a General Electric (GE) fuel rod assembly and tie plate and against correlations based on results from small-scale, air-water tests of a perforated plate conducted at Northwestern University. Sensitivity studies showed that variation of the critical parameters in the code had no significant influence on predicted peak cladding temperature over the limited range explored.

WCOBRA/TRAC was constructed out of numerous models and correlations derived from limited tests at facilities that often differ greatly from full-scale PWRs (e.g., air-water tests in small, long, straight pipes at low pressure). Many of the correlations, formulae, and models are particularly suspect in the UPI context. For example:

- The physical models in the code are not particularly good for predicting two-phase flows in straight pipes. It is truly remarkable that these same models are able to come so close to representing CCFL data for GE tie plates modeled as an effective length of straight pipe.

- The nodalization used by Westinghouse results in modeling of favorable paths for water penetration to the lower plenum. This, however, is only an approximate treatment of the many parallel paths provided by the numerous holes in the tie plate. Such problems have been addressed more comprehensively in the chemical industry.

- No attempt is made to realistically model the thermal-hydraulic phenomena in the upper plenum. A jet with considerable momentum, directed at a forest of structures, is treated as either a slug of water with no momentum or as a dispersed fog of drops with no momentum. Both assumptions are unrealistic, and it is not conclusive that they bound the actual behavior, but they are used as the basis for sensitivity calculations.

- The model for de-entrainment in WCOBRA/TRAC is based on droplet diffusion, not on the inertial impaction that actually occurs.

- Condensation is empirically modeled by means of a coefficient, which Westinghouse varies over a limited range, that does not reflect basic technical uncertainty and is tuned to a small data set.

The staff has stated that the CSAU evaluation methodology was followed, but we recognize a number of shortcomings:

- CCFL modeling was verified from the GE tests but data from separate-effects tests performed at the University of Hanover and at the Idaho National Engineering and Environmental Laboratory, using PWR geometries that are more typical of Westinghouse plants, were ignored. CCFL is known to be significantly dependent on geometrical details.

- Results of the UPTF tests show that more condensation occurred in the upper plenum than was predicted by the code. Yet, the condensation coefficient was not ranged upwards to try to represent this. Had this been done, the predicted CCFL would probably have been more restrictive.

- There was little investigation of the possibility of compensating errors. For example, the underestimation of condensation mentioned above was probably balanced by underestimation of three-dimensional effects that allowed more penetration of water than was permitted by the limited noding in the code formulation.

- Although the calculated peak cladding temperature was insensitive to variations in the parameters that were ranged, it is clear that there are some values for these parameters, particularly interphase drag, that would significantly restrict water penetration. It would have been useful to extend the exploration of parameters into this region in order to know how much margin was available in the uncertainty range for coefficients that are known to be sensitive to conditions such as geometrical details.

We and our consultants raised these and other technical issues during our discussions, but Westinghouse and the staff regarded them as irrelevant to the overall conclusions. Although this may be broadly true in the present context, there is no assurance that it will always be so. Therefore, we believe that the staff needs to provide more explicit guidance regarding the quality of the application of the CSAU evaluation methodology and the code validation requirements. The lessons learned during this review are particularly timely because the staff is presently developing such guidance for future code evaluations. We believe that to carry out such evaluations the staff should:

- Have the capability to run the codes under review in a comprehensive, probing, critical, and objective manner so that a truly independent assessment is made.

- Maintain a thorough understanding of technical issues so that it is aware of when to question circumstances in which codes may be misleading or inadequate. One cannot rely on assurances from protagonists or on a routine following of steps in a process.

- Have its own code of sufficient quality that it can be used to assess the viability of other codes in situations where experimental evidence is not available or is inconclusive.

Throughout the coming year, we will be reviewing other codes intended for use in safety analyses. We look forward to working with the staff to develop the appropriate procedures.

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

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

<u>References</u>:

1. Letter dated August 6, 1998, from H. A. Sepp, Westinghouse, to Nuclear Regulatory Commission, transmitting Comparison of Best-Estimate LOCA Methodologies for Westinghouse PWRs With Upper Plenum Injection and Cold Leg Injection.

2. Westinghouse Topical Report, WCAP-14449-P, "Application of Best Estimate Large Break LOCA Methodology to Westinghouse PWRs With Upper Plenum Injection," August 1995, including an appendix of information provided to the NRC in response to requests for additional information on WCAP-14449-P **(contains proprietary information).**

3. Letter dated December 2, 1998, from H. A. Sepp, Westinghouse, to Nuclear Regulatory Commission, Subject: Information Regarding the December 16, 1998, Meeting With the ACRS Thermal-Hydraulic Phenomena Subcommittee.

4. Excerpts from Westinghouse Topical Report, WCAP-12945-P-A, "Westinghouse Code Qualification Document for Best Estimate Loss of Coolant Accident Analysis," March 1998 **(contains proprietary information).**

5. Letter dated July 12, 1995, from N. J. Liparulo, Westinghouse, to Nuclear Regulatory Commission, Subject: Summary of Westinghouse Best-Estimate LOCA Methodology.

6. E-Mail dated February 16, 1999, from G. Wallis, ACRS Member, to P. Boehnert, ACRS Staff, transmitting list of questions for Westinghouse response at the February 23, 1999 Thermal Hydraulic-Phenomena Subcommittee Meeting.

7. Response from Westinghouse to G. Wallis, ACRS Member, regarding List of Questions to be addressed at the February 23, 1999 Thermal-Hydraulic Phenomena Subcommittee Meeting.

8. Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Acceptability of the Westinghouse Topical Report WCAP-14449(P), "Application of Best-Estimate Large Break LOCA Methodology to Westinghouse PWRs With Upper Plenum Injection."

9. U. S. Nuclear Regulatory Commission report prepared by Idaho National Engineering and Environmental Laboratory, INEEL/EXT-98-00802, Draft, Rev. 1, "Draft Technical Evaluation Report, Application of Best-Estimate Large Break LOCA Methodology to Westinghouse PWRs With Upper Plenum Injection, WCAP-14449-P," undated.

10. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance," dated May 1989.

11. U.S. Nuclear Regulatory Commission Report, NUREG/CR-5249, "Quantifying Reactor Safety Margins - Application of the Code Scaling, Applicability, and Uncertainty Evaluation Methodology to a Large-Break, Loss-of-Coolant Accident," December 1989.

12. ACRS Report dated February 23, 1996, from T. S. Kress, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: Westinghouse Best-Estimate Loss-of-Coolant Accident Analysis Methodology.

13. ACRS Report dated April 19, 1996, from T. S. Kress, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: Westinghouse Best-Estimate Loss-of-Coolant Accident Analysis Methodology.