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Mr. Victor Stello, Jr.
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Stello:

SUBJECT: COMMENTS ON THE NRC STAFF'S DRAFT SAFETY EVALUATION OF THE WESTINGHOUSE TOPICAL REPORT, WCAP-10924, "WESTINGHOUSE LARGE-BREAK LOCA BEST-ESTIMATE METHODOLOGY"

During the 339th meeting of the Advisory Committee on Reactor Safeguards, July 14-16, 1988, we met with members of the NRC Staff and their consultants as well as representatives of the Westinghouse Electric Corporation and licensees of the Prairie Island and Point Beach nuclear power plants and reviewed the subject safety evaluation. Our Subcommittee on Thermal Hydraulic Phenomena met on May 27 and June 21, 1988 to discuss this matter. We also had the benefit of the documents listed as references for this letter.

This review concerns an improved method for predicting the performance of the emergency core cooling system (ECCS) during a large-break loss-of-coolant accident (LOCA) in two-loop pressurized water reactor (PWR) plants of Westinghouse design. The ECCS in these plants is different from those in the majority of PWR plants in that low-pressure injection flow is provided to the reactor vessel through special nozzles entering the upper plenum rather than through the cold leg piping.

Several years ago, the NRC Staff recommended that certain thermal-hydraulic conditions unique to the upper plenum injection (UPI) configuration were not adequately modeled in the ECCS codes being used. Accordingly, licensees at UPI plants were instructed to develop improved models, to demonstrate that these models provided compliance with the ECCS rule, and to use these models in future licensing submittals. Westinghouse has developed such an improved model for the utilities operating the Prairie Island and Point Beach nuclear power plants. That is the subject of the present review. Models for two other UPI plants are being developed elsewhere and are to be given a separate review.

The improved analysis is provided by a code called WCOBRA/TRAC that has been developed from existing codes TRAC-PD2 and COBRA-TF. COBRA-TF is the reactor vessel part of the analysis while TRAC-PD2 models the overall system. COBRA/TF describes three flow fields in three dimensions. It thus has the capability of modeling thermal-hydraulic phenomena important in the UPI configuration, including details of counter-current flows of vapor and liquid.

In addition to the improved phenomenological modeling, the overall methodology presented in WCAP-10924 incorporates a so-called "best-estimate" approach to calculation of limiting plant parameters. The general approach described in the referenced NRC document, SECY-83-472, is used.

The staff's review has concluded that the WCOBRA/TRAC code provides adequate modeling to represent the UPI plant configuration for large-break LOCA analysis. They have also concluded that the best-estimate methodology, including allowances for uncertainty, adequately conform to the provisions outlined in SECY-83-472. However, they have not accepted the best-estimate methodology, as presented, for other plants or for use with the proposed new ECCS rule when that becomes available. We find no reason to disagree with these conclusions of the staff.

A cautionary word about the so-called "best-estimate" approach: we have previously expressed our approval of, and, in fact, applauded, this approach to analysis of reactor transients. This applies to both the SECY-83-472 approach and the proposed new rule for large-break LOCA analysis. Best-estimate analysis, in this sense, has two parts: (1) a realistic analysis, with no purposeful biases, to provide a central estimate of the parameters of interest and (2) a conscious and explicit estimate of the margin that should be provided from this central estimate to achieve a desired level of confidence in conclusions to be drawn from the analysis.

In most practical engineering situations, including LOCA analysis, the relationships are so complex and the data so sparse that mathematical rigor in defining the desired confidence level and necessary allowance is impossible. However, the method is still of value even though it may involve what are largely engineering judgments about confidence level and the magnitude of allowances. The problem is that, too often, practitioners of the best-estimate analysis or users of the results describe their analysis and the results in terms that imply mathematical rigor and give an impression that statistical relationships have been developed with great precision, whereas, the actual data and methods of analysis are approximate. Terms such as "95% confidence interval" are used when only a term such as "a reasonably high confidence level" is justified. In addition, distinctions among variability, uncertainty, and confidence level are not observed, and statistical relationships are often used carelessly and inaccurately. We recommend that, in the future, the NRC Staff should involve professional statisticians in the review of these matters.

Also, there will be a greater technical challenge for the staff in reviewing best-estimate analyses compared with evaluation-model reviews carried out in the past. We believe that agency management will have to make a special effort to provide appropriate resources.

We hope that our comments will prove useful.

Sincerely,

W. Kerr
Chairman

References:

1. Memorandum dated July 12, 1988, from M. W. Hodges, NRC, to P. Boehnert, NRC, transmitting draft "Safety Evaluation of the

Westinghouse Electric Corporation Topical Report, WCAP-10924, 'Westinghouse Large-Break LOCA Best-Estimate Methodology'" (Proprietary)

2. U.S. Nuclear Regulatory Commission Staff Document, "Emergency Core Cooling System Analysis Methods," SECY-83-472, dated November 17, 1983
3. Westinghouse Electric Corporation, WCAP-10924-P: "Westinghouse Large Break LOCA Best Estimate Methodology - Volume 1: Model Description and Validation," June 1986 (Proprietary); and WCAP-10924-P, Volume 2, Revision 1: "Application To Two-Loop PWRs Equipped With Upper Plenum Injection," April 1988 (Proprietary)
4. Westinghouse Electric Corporation: "Responses to NRC Questions on Westinghouse Large Break LOCA Best Estimate Methodology, WCAP-10924-P, Volume 1," October 1987 (Proprietary)
5. Westinghouse Electric Corporation: WCAP-10924-P, Volume 2, Addendum 1: "Responses to NRC Questions on WCAP-10924-P, Volume 2 (Addendum to Westinghouse Large Break LOCA Best-Estimate Methodology, Volume 2: Application to Two-Loop PWRs Equipped With Upper Plenum Injection)," April 1988 (Proprietary)
6. Letter dated June 14, 1988, from L. E. Hochreiter, Westinghouse to V. E. Schrock, University of California, Berkeley, transmitting excerpt of Westinghouse Report: "Appendix B Heat Source Models" (undated)
7. Westinghouse Electric Corporation, WCAP-10924-P, Volume 1, Addendum 1: "Westinghouse Large-Break LOCA Best Estimate Methodology, Volume 1: Model Description and Validation (Addendum 1: Revised Code Predictions and Code Uncertainty)," April 1988
8. Letter dated June 15, 1988, from L. E. Hochreiter, Westinghouse to I. Catton, University of California, Los Angeles, transmitting WCAP-8951: "Method of Analysis and Evaluation of Jet Impingment Loads from Postulated Pipe Breaks," (undated); and WCAP-9748: "Westinghouse Owners Group Asymmetric LOCA Loads Evaluation Phase C (NRC TAP-Topic A-2)," June 1980
9. Letter dated June 30, 1988, from L. E. Hochreiter, Westinghouse to D. DiIanni, NRC, transmitting Westinghouse Letter Report: "CCTF Four Channel Core Analysis - Section 1" (Proprietary - Undated); Westinghouse Report: "Supplemental Information on the Calculation of W COBRA/TRAC Uncertainties From WCAP-10924, Volume 1, Revision 2," June 1988; and WCAP-10924-P, Volume 1, Addendum 2: "Westinghouse Large-Break LOCA Best-Estimate Methodology Volume 1: Model Description and Validation (Addendum 2: Revised Appendix B: Heat Source Models)," July 1988
10. Letter dated June 15, 1988, from L. E. Hochreiter, Westinghouse to I. Catton, University of California, Los Angeles, transmitting JAERI-Memo 60-142: "Quick-Look Report on CCTF Core-II Reflood Test CS-16 (Run 76) - Effect of Asymmetric Upper Plenum Injection on Reflood Phenomena," Japan Atomic Energy Research Institute, June 1985

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