United States Nuclear Regulatory Commission
Office of Public Affairs
Washington, DC 20555
Phone 301-415-8200 Fax 301-415-2234
Internet:opa@nrc.gov

No. 96-44 FOR IMMEDIATE RELEASE
(Tuesday, February 27, 1996)

NOTE TO EDITORS:

The Nuclear Regulatory Commission has received the attached report from its Advisory Committee on Reactor Safeguards. The report, in the form of a letter, provides comments on a new method developed by the Westinghouse Electric Corporation for evaluating a large break, loss-of-coolant accident.

In addition, the NRC's Executive Director for Operations received a letter concerning a revision to NRC's Regulatory Guide 1.149, "Nuclear Power Plant Simulation Facilities for Use in Operator License Examinations."

#

Attachments:
As stated

February 23, 1996

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

Dear Chairman Jackson:

SUBJECT: WESTINGHOUSE BEST-ESTIMATE LOSS-OF-COOLANT ACCIDENT ANALYSIS METHODOLOGY

During the 428th meeting of the Advisory Committee on Reactor Safeguards, February 8-10, 1996, we reviewed the best-estimate, large-break, loss-of-coolant accident (LBLOCA) analysis methodology developed by the Westinghouse Electric Corporation. During this review, we had the benefit of discussions with representatives of the NRC staff, Westinghouse, Idaho National Engineering Laboratory, and several nuclear power plant licensees. Our Subcommittee on Thermal Hydraulic Phenomena has held a number of meetings on this matter as far back as 1991.

The last meeting of the Subcommittee concerning this issue was held on January 18-19, 1996. We also had the benefit of the referenced documents.

Westinghouse has developed an improved method to evaluate the performance of emergency core cooling systems (ECCS) for the case of a LBLOCA in three- and four-loop pressurized-water reactors (PWRs) of Westinghouse design. Westinghouse has proposed that this improved method, based on the use of the WCOBRA/TRAC code, be accepted for routine use in demonstrating that the cores in these plants meet NRC licensing requirements pursuant to the revised ECCS Rule (10 CFR 50.46). The NRC staff has reviewed this proposal and has concluded that the new methodology can be used for licensing calculations. We concur with the staff; however, some improvements in the uncertainty analysis are desirable.

The improved method of analysis takes advantage of data and the understanding of thermal-hydraulic behavior developed during the past two decades. This method will reduce the conservative margins in the calculated peak cladding temperature that result from the use of current methods based on Appendix K. This will permit licensees of Westinghouse three- and four-loop PWRs to have greater flexibility in the operation of their plant reactor cores and in associated fuel management practices. We also believe that, when properly documented, the improved method will provide a straightforward and understandable assessment of the performance of an important safety system.

The improved LOCA evaluation method makes use of realistic values for inputs and correlations rather than the conservatively biased values used in the past. To meet licensing requirements, empirically based uncertainty distributions for each of the important inputs and correlations are used and propagated through the solution algorithm, WCOBRA/TRAC, to obtain estimates of uncertainty distributions for the peak cladding temperature. A nominal 95 percent probability of nonexceedance is required for licensing purposes. Questionable models or correlations are adjusted to ensure that their predictions are conservative. Westinghouse expanded the Code Scaling, Applicability, and Uncertainty (CSAU) evaluation methodology outlined in NRC Regulatory Guide 1.157, by including additional parameters not considered during the earlier CSAU exercise conducted by the NRC staff.

We have some concerns about the Westinghouse best-estimate LBLOCA evaluation methodology. The method used by Westinghouse to obtain the heat transfer coefficient uncertainty distribution resulted in some high values that are nonphysical. Westinghouse should reevaluate the heat transfer uncertainty distribution with appropriate consideration of the dependencies on physical parameters such as reflood rate. The Westinghouse treatment of the minimum wetting (or rewetting) temperature is not satisfactory

because the correlation ignores important phenomena and could lead to nonconservative results. The existence of compensating errors in WCOBRA/TRAC may be a reason for the skewed distribution in heat transfer coefficients. We believe that these concerns should be addressed.

Obtaining adequate documentation in a timely manner has been a problem from the outset of this review. This has unnecessarily complicated the reviews by both the NRC staff and the ACRS. Westinghouse has committed to provide documentation that will clearly lay out its LBLOCA methodology. We believe that the staff should review this final documentation prior to approving use of the improved methodology. The staff should also prepare guidelines for documentation of future best-estimate LOCA submittals before the lessons learned from this review are forgotten.

It is important to realize that the deficiencies seen in codes like TRAC and RELAP may preclude their extension to the evaluation of best-estimate ECCS performance under small-break LOCA conditions or to passive plant designs. The use of WCOBRA/TRAC is acceptable for LBLOCA calculations because of the extensive test data available for code validation and the associated analytical expertise developed over the past 20 years. A comparable database does not exist for many other applications.

We commend the staff and Westinghouse for completing an important task. If the above concerns are adequately addressed, the result will be a much improved best-estimate method for the prediction of LBLOCA behavior in light-water reactors.

ACRS Member George Apostolakis did not participate in the Committee's deliberation of this matter.

Sincerely,

/s/

T. S. Kress
Chairman, ACRS

References:

1. Westinghouse Topical Report, "Code Qualification Document for Best Estimate LOCA Analysis," WCAP-12945-P, Revision 1, Volumes 1-5, June 1992 (Proprietary)
2. U. S. Nuclear Regulatory Commission, "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Acceptability of the Topical Report WCAP-12945(P) 'Westinghouse Code Qualification Document for Best-Estimate Loss of Coolant Accident Analysis' for Referencing in PWR Licensing Applications, Westinghouse Electric Corporation" (Draft) and "Draft Technical Evaluation Report, Westinghouse

Code Qualification Document for Best Estimate Loss of Coolant Accident Analyses, WCAP-12945-P" (Proprietary), transmitted by P. Boehnert, ACRS staff, to the ACRS Thermal Hydraulic Phenomena Subcommittee, by memorandum dated January 4, 1996

3. Memorandum, dated November 3, 1995, from P. Boehnert, ACRS staff, to I. Catton, Chairman, ACRS Thermal Hydraulic Phenomena Subcommittee, Subject: NRC/NRR-Westinghouse Meeting, October 23-24, 1995 - Westinghouse Best-Estimate ECCS Evaluation Model Code, WCOBRA/TRAC", including W memorandum, dated October 13, 1995, transmitting "Revisions to the W Best-Estimate Uncertainty Methodology" (Proprietary)
4. Memorandum dated January 5, 1996 from M. Nissley, Westinghouse, to Members and Consultants of the ACRS Thermal Hydraulic Phenomena Subcommittee, transmitting the following reports:

   - NTD-NRC-95-4505 - Roadmap Comparison with CSAU Methodology
   - NTD-NRC-95-4575 - Revised Uncertainty Methodology Report (Proprietary)
   - NTD-NRC-95-4586 - Assessment of Compensating Errors (Proprietary)
   - NTD-NRC-95-4588 - Non-Proprietary Executive Summary
   - NTD-NRC-96-4618 - Responses to Several Issues Identified in INEL's Review of NTD- NRC-95-4575 (Proprietary)

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

February 11, 1996

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

Dear Mr. Taylor:

SUBJECT: REVISION 2 TO REGULATORY GUIDE 1.149, "NUCLEAR POWER PLANT SIMULATION FACILITIES FOR USE IN OPERATOR LICENSE EXAMINATIONS"

During the 428th meeting of the Advisory Committee on Reactor Safeguards, February 8-10, 1996, we heard presentations by and held discussions with representatives of the NRC staff and the Nuclear Energy Institute concerning Revision 2 to Regulatory Guide 1.149. We also had the benefit of the documents referenced.

This revision to the Regulatory Guide describes a method acceptable to the NRC staff for complying with those portions of 10 CFR Part 55, "Operators' Licenses," that relate to the use of simulation facilities in the licensing of nuclear power plant operators. The current version of this Regulatory Guide endorses ANSI/ANS-3.5-1985, "Nuclear Power Plant Simulators for Use in Operator Training and Examinations," with some clarifications and exceptions. Revision 2 to the Regulatory Guide endorses ANSI/ANS-3.5-1993, again with some clarifications and exceptions. The NRC staff has met with industry representatives, including representatives of the ANSI/ANS-3.5 Working Group, to discuss the proposed Revision 2 to the Regulatory Guide and has considered industry comments in the proposed final version.

We believe that the staff should proceed with the publication of this Regulatory Guide to be consistent with the current state of the art with respect to the use of nuclear power plant simulators.

Sincerely,

/s/

T. S. Kress
Chairman, ACRS

References:

1. Memorandum dated January 30, 1996, from Bill M. Morris, Office of Nuclear Regulatory Research, NRC, to John T. Larkins, ACRS, Subject: Proposed Resolution of Draft Regulatory Guide DG-1043, Nuclear Power Plant Simulation Facilities for Use in Operator License Examinations
2. American Nuclear Society, ANSI/ANS-3.5-1993, "Nuclear Power Plant Simulators for Use in Operator Training and Examination," March 29, 1993