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FOR IMMEDIATE RELEASE
(Tuesday, April 18, 1995)

NOTE TO EDITORS:

The Nuclear Regulatory Commission has received two reports from its independent Advisory Committee on Reactor Safeguards. The attached reports, in the form of letters, comment on proposed rulemaking on reporting reliability and availability information for risk-significant systems and equipment and a proposed final rule change to technical specifications for licensed nuclear power plants.

In addition, the NRC's Executive Director For Operations has received a letter report from the ACRS on the NRC's test and analysis program in support of the design certification review for the Westinghouse AP600 advanced light water reactor.

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Attachments:
As stated

April 13, 1995

The Honorable Ivan Selin, Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Selin:

SUBJECT: PROPOSED FINAL RULE CHANGE TO 10 CFR 50.36, TECHNICAL SPECIFICATIONS

During the 420th meeting of the Advisory Committee on Reactor Safeguards, April 6-7, 1995, we discussed with representatives of the NRC staff and the Nuclear Energy Institute the subject proposed final rule change to technical specifications. We had the benefit of the documents listed.

The "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors," dated July 23, 1993, established four criteria to define requirements that should be controlled by technical specifications. The Commission concluded that it was appropriate to codify these criteria in a rule that would be consistent with the Policy Statement and preserve the voluntary nature of adopting the improved Standard Technical Specifications for previously licensed plants.

In our June 18, 1993 report, we stated our agreement with the views expressed by the Commission on this matter and concluded that the staff had appropriately modified the Policy Statement in response to the Commission's comments. We did express a concern that there was a need for more detailed guidance on the definition of "significant to public health and safety" as it is used in Criterion 4 of the final Policy Statement.

The staff proposes to implement Criterion 4 in a manner consistent with the Commission's policies on the use of probabilistic risk assessment methods and the staff's PRA Implementation Plan.

The staff maintains that the improved Standard Technical Specifications, the final Policy Statement, the Backfit Rule, and the statement of consideration for this proposed final rule change contain sufficient guidance for implementing Criterion 4. We do not agree with this position.

We have previously objected to regulations that are subject to a variety of interpretations which rely solely on the judgment of the regulator. In the interest of coherence in regulation and predictability of the regulatory process, we recommend that codification of the rule include more explicit definition and guidance on the implementation of the "significant to public health and safety" provision of Criterion 4. We believe a rule that omits this is not complete and will not meet the pressing need for a rule on Technical Specifications Improvements. We recommend delaying issuance of the rule until it is complete.

Sincerely,

T. S. Kress
Chairman, ACRS

References:
1. Draft Commission Paper, from James M. Taylor, Executive Director for Operations, for the Commissioners, Subject: Final Rulemaking Package for 10 CFR 50.36, "Technical Specifications," (Predecisional) transmitted by Memorandum dated March 27, 1995, from B. K. Grimes to John T. Larkins
2. Staff Requirements Memorandum dated May 25, 1993, from Samuel J. Chilk, Secretary, for James M. Taylor, Executive Director for Operations, Subject: SECY-93-067 - Final Policy Statement on Technical Specifications Improvements
3. ACRS letter dated June 18, 1993, from J. Ernest Wilkins, Jr., ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: Policy Statement on Technical Specifications Improvements for Nuclear Power Plants
4. Nuclear Regulatory Commission, 10 CFR Part 50, Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors, July 23, 1993
5. SECY-94-219 dated August 19, 1994, from James M. Taylor, Executive Director for Operations, for the Commissioners, Subject: Proposed Agency-Wide Implementation Plan for Probabilistic Risk Assessment (PRA)
6. Nuclear Regulatory Commission, "Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities: Proposed Policy Statement," issued for public comment on December 1, 1994

April 12, 1995

The Honorable Ivan Selin, Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Selin:

SUBJECT: PROPOSED RULEMAKING ON REPORTING RELIABILITY AND AVAILABILITY INFORMATION FOR RISK-SIGNIFICANT SYSTEMS AND EQUIPMENT

During the 419th and 420th meetings of the Advisory Committee on Reactor Safeguards, March 9-10 and April 6-7, 1995, we discussed with representatives of the NRC staff and the Nuclear Energy Institute a proposed rule that would require licensees to report reliability and availability data for risk-significant systems and equipment. We also had the benefit of the documents listed.

Data on the reliability and availability of risk-significant systems and equipment are essential for the expanded use of risk-based regulation. Plant-specific data could augment the effectiveness and efficiencies attributed to risk-based regulation. Neither the Licensee Event Reports nor the Nuclear Plant Reliability Data System provide all the data that are needed to support risk-based regulation.

The proposed rule would require licensees to provide periodic summary reports to the NRC on reliability and availability data for risk-significant systems and equipment. Records and analyses of demands, failures, and unavailabilities that provide the bases for these summary reports would be maintained onsite and would be available for NRC inspection.

The regulatory analysis developed by the staff indicates that a reliability and availability data base would provide significant benefits to both the licensees and the NRC. As part of the implementation of the Maintenance Rule, licensees will be required to maintain records of most, if not all, of these reliability and availability data. The staff plans to issue a final rule and its associated guidance document at the same time the Maintenance Rule goes into effect.

Representatives of the staff, the Institute of Nuclear Power Operations, and the Nuclear Energy Institute have reached agreements on the risk-significant systems and equipment that need to be addressed in the availability and reliability data base. The needed data on these systems and equipment have been defined. The staff is now proposing pilot programs to continue refinement of these definitions and to demonstrate the utility of the data base.

The staff feels that availability and reliability data needed to support risk-based regulation should be publicly available. The licensees have, however, taken the position that they will not voluntarily submit data on reliability and availability if it is to become public.

We believe that high-quality, plant-specific reliability and availability data are needed if risk-based regulation is to fully reach its potential for both improving safety and reducing burdens on licensees. Our view on the public availability of the data is that the staff has taken the correct position. Consequently, we recommend publication of the proposed rule for public comment. We believe that the public comment process will be greatly enhanced if, at scheduled workshops, the staff presents examples of how data on reliability and availability will be applied.

Sincerely,

T. S. Kress
Chairman, ACRS

References:

1. U.S. Nuclear Regulatory Commission, Draft Regulatory Analysis dated March 31, 1995, Subject: Reporting Reliability and Availability Information for Risk-Significant Systems and Equipment (received April 3, 1995) (Predecisional)
2. U.S. Nuclear Regulatory Commission, Draft 10 CFR Part 50, RIN 3150-AF33, "Reporting Reliability and Availability Information for Risk-Significant Systems and Equipment" (received April 3, 1995) (Predecisional)
3. Memorandum dated October 4, 1994, from Edward L. Jordan, Office for Analysis and Evaluation of Operational Data, to James M. Taylor, NRC Executive Director for Operations, Subject: Rulemaking to Collect Safety/Risk-Significant System and Equipment Reliability/Availability Data

April 12, 1995

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 PROGRAM IN SUPPORT OF AP600 ADVANCED LIGHT WATER PASSIVE PLANT DESIGN REVIEW

During the 420th meeting of the Advisory Committee on Reactor Safeguards, April 6-7, 1995, we discussed the confirmatory test and analysis program being conducted by the Office of Nuclear Regulatory Research (RES) in support of the design certification review for the Westinghouse AP600 advanced light water reactor. During this meeting, we had the benefit of discussions with representatives of RES and its contractor, the Idaho National Engineering Laboratory. Our Subcommittee on Thermal Hydraulic Phenomena held a meeting on March 27-28, 1995, to discuss this matter. The Committee previously reviewed this matter during its October and November 1994 meetings and provided formal comments in its November 10, 1994 letter. We also had the benefit of the documents listed.

During the past year, the RES thermal-hydraulic program has undergone a dramatic change for the better. The presentations made to the Thermal Hydraulic Phenomena Subcommittee and the Committee were clear, well-organized, and demonstrated good technical thinking. We compliment the management, the staff, and the contractors for the improvement. We also note that RES is making good use of a cadre of high-quality thermal-hydraulic consultants.

Completion of the Phenomena Identification and Ranking Table (PIRT) for the AP600 remains an important task. It was much easier to develop the PIRT for the current operating plants because a great deal of relevant test data were available. This is not the case for the AP600 and SBWR passive plants. Development of the PIRT should be concurrent with a scaling analysis and review of test results to provide quantitative support for the engineering judgments that must be made. The RES approach appears to be systematic and well organized. We recommend, however, that RES fully document the development of the PIRT.

The RES analysis of test data from ROSA and Oregon State University (OSU) was very thorough. We encourage the staff to continue such efforts, while drawing on the insights from the ongoing scaling analysis. RES should strive to provide complete documentation of the test analysis effort and should also document the phenomena that are not important.

The ongoing RES scaling analysis for the test facilities is an important effort. This analysis can be used to assess the impact of scaling distortions and atypicalities of the different facilities to support the conclusions of PIRT as well as to understand the physical phenomena important to AP600 thermal-hydraulic behavior. For the current operating plants, the PIRT was developed for existing systems whose thermal-hydraulic behavior was demonstrated over a 20-year period. For the AP600 design certification review, however, comparable understanding must be gained quickly. We believe that a rigorous analysis of test data based on the use of a good scaling analysis and the PIRT should permit this to be done. We recommend that the OSU scaling effort performed in support of the Westinghouse test program be a starting point for the development of a consistent set of AP600 scaling criteria for application to the ROSA, OSU, and SPES test facilities.

Several issues were discussed during our meetings with RES. The first is the potential for water hammer in the AP600 design during LOCAs. Attention should be given to identifying where and under what circumstances water hammer could occur. A second is the potential for thermal stratification in the Core Makeup Tank, the Incontainment Refueling Water Storage Tank, and in the horizontal pipe runs of the reactor coolant system. The occurrence of thermal stratification in the cold leg combined with the possibility of steam injection could be a precursor to a significant water hammer. We recommend that the potential safety problems caused by these phenomena be identified and their significance to safety be assessed soon in order to avoid questions at the time of certification. The RES thermal-hydraulic consultants could be very helpful in this regard.

We are concerned about the applicability of the present thermal-hydraulic codes (TRAC, RELAP5) for analysis of plants like the AP600. These codes have to predict types of thermal-hydraulic behavior for which they have been shown to be weak; i.e., prediction of condensation, thermal stratification, and water level. We recommend that RES consider developing a contingency plan in the event that the codes cannot adequately predict these key phenomena.

Although the focus of our meetings with RES was on the development of the PIRT, some reference was made to determination of computational uncertainty. The uncertainty parameter of choice is peak clad temperature for the large-break LOCA while reactor vessel primary system inventory is the choice for the small-break LOCA. With resources being reduced, we recommend that RES focus its attention on the more safety-significant small-break LOCA.

Overall, much progress in the RES thermal-hydraulic program is evident. It is well structured and will yield a great deal of valuable insight into the behavior of passive plants.

Sincerely,

T. S. Kress
Chairman, ACRS

References:

1. Memorandum dated February 14, 1995, from M. Wayne Hodges, NRC Office of Nuclear Regulatory Research, to John T. Larkins, ACRS Executive Director, transmitting INEL draft report, "Interim Phenomena Identification and Ranking Tables for Westinghouse AP600 Small Break Loss-of-Coolant Accident, Main Steam Line Break, and Steam Generator Tube Rupture Scenarios," INEL-94/0061
2. Memorandum dated February 14, 1995, from M. Wayne Hodges, NRC Office of Nuclear Regulatory Research, to John T. Larkins, ACRS Executive Director, transmitting LANL draft report by B. Boyack, "AP600 Large-Break Loss-of-Coolant Accident Phenomena Identification and Ranking Tabulation"
3. Letter dated February 15, 1995, from Gary E. Wilson, INEL, to Tim Lee, NRC, Subject: Transmittal of AP600 T/H Consultants Meeting Minutes
4. ACRS report dated November 10, 1994, from T. S. Kress, ACRS Chairman, to James M. Taylor, Executive Director for Operations, Subject: NRC Test and Analysis Programs in Support of AP600 and SBWR Advanced LWR Passive Plant Design Certification Reviews