No. 95-80 FOR IMMEDIATE RELEASE
Tel. 301/415-8200 (Thursday, June 22, 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:

1) A proposed final policy statement on the use of probabilistic risk assessment methods in nuclear regulatory activities; and

2) Proposed final revisions to Appendix J of the NRC's Part 50 regulation, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors."

In addition, the NRC's Executive Director for Operations has received two ACRS letter reports. They comment on:

1) A proposed final rule and regulatory guide for fracture toughness requirements for light water reactor pressure vessels; and

2) A proposed Commission paper on staff positions on technical issues pertaining to the Westinghouse standardized nuclear reactor design designated AP600.

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Attachments:
As stated

June 15, 1995

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

Dear Mr. Taylor:

SUBJECT: PROPOSED COMMISSION PAPER ON STAFF POSITIONS ON TECHNICAL ISSUES PERTAINING TO THE WESTINGHOUSE AP600 STANDARDIZED PASSIVE REACTOR DESIGN

During the 422nd meeting of the Advisory Committee on Reactor Safeguards, June 8-10, 1995, we discussed the subject Commission paper. Our Subcommittee on Westinghouse Standard Plant Designs met on May 31, 1995, to review this matter. During these meetings, we had the benefit of discussions with representatives of the staff and Westinghouse. We also had the benefit of the documents referenced.

The intent of the proposed Commission paper is to record the staff positions on ten separate issues. In some cases, however, the reviews have not progressed to the point that the staff can recommend a position. In such cases, the paper describes the approach that Westinghouse is proposing in its application with little staff comment. The staff is continuing its review of these matters.

Our comments follow the same organization found in the attachment to the paper.

I. <u>Leak-Before-Break Approach</u>

Westinghouse proposes that any dynamic effects associated with postulated pipe ruptures in a broad range of pipe sizes can safely be excluded from the AP600 piping design basis by virtue of the current understanding of leakage and flaw sizes, and the proposed leakage rate limit of 0.5 gpm. The range of pipe sizes (4 inch diameter and greater) that would be covered by the leak-before-break (LBB) approach is broader than that allowed in currently operating pressurized water reactors for which the usual plant leakage rate limit is set at 1.0 gpm.

The staff agreed that the leakage rate limit of 0.5 gpm is achievable in the AP600 design but wishes to add conservatism in applying the LBB approach at the design certification stage by requiring that all loads used in the piping design be multiplied by a factor of 1.4. The staff considers this prudent because the detailed design of piping configuration and the as-built stress levels will not be available for review at the certification stage. Westinghouse argued that this added conservatism is not needed and will act to limit the gains in plant arrangement, economy, and safety that application of the LBB approach could provide.

We believe that the staff is hard pressed to justify adding conservatism on all the piping loads above that which has been applied to other plants. Although it is true that the details of the piping design are some years away, the staff and Westinghouse should now be able to combine the standard piping design protocols with what is known about the performance of flawed pipes into a design criterion without excessive conservatism.

II. Security Design

The proposed AP600 plant arrangement includes a vehicle barrier at a "stand-off distance," but the personnel access control will is located within the nuclear island of the plant. The vital areas of the plant are coterminous. This feature is not specific to the passive nature of the plant design and might be offered in other plant designs as well. The staff continues to review the proposed design, but seems receptive to the idea. The staff believes that inspections, tests, analyses, and acceptance criteria (ITAAC) may be required for this security design.

We believe the proposed security design could meet the safety and security requirements when implemented, and we are interested in the continuing staff review of the proposed design. We also noted that the design seems to offer less flexibility for the many work access points that operating plants need during outage periods.

III. Technical Specifications

Westinghouse proposes that hot shutdown, rather than cold shutdown, be considered the safe shutdown end state. The staff evaluation has not progressed to the point where the staff could make substantial comment. We also will withhold comment at this time. We expect that review of the probabilistic risk assessment regarding this issue will be instructive.

IV. Initial Test Program

Westinghouse and the staff have been discussing the content of the initial test program to be performed by the first plant built under the design certification, and test programs to be performed by subsequent plants. We believe that the staff is approaching the matter appropriately. When the discussions have resulted in new submittals from Westinghouse, we may have more information on which to comment.

V. Passive System Thermal-Hydraulic Performance Reliability

The staff believes that the magnitude of the natural forces relied on for the passive safety systems leads to large uncertainties in the thermal-hydraulic performance. It stated that one could quantify these uncertainties, but only with "a prohibitively large number of computations." The staff proposed instead that a surrogate conservative risk-based margins approach be developed to eliminate the need to quantify thermal-hydraulic uncertainty for most, if not all, accident sequences.

This approach may be expedient, but we believe efforts should continue on the quantification of the uncertainty for use in probabilistic risk assessments.

VI. Regulatory Treatment of Non-Safety Systems

Westinghouse and the staff have been meeting to review the need for some level of regulatory treatment for systems and components that are not safety grade, but that have important support and backup functions. A key issue identified by the staff in this regard is the reliance that Westinghouse places on equipment or materials that may be required beyond 72 hours following an accident but which are not to be stored onsite. The staff review of this issue is currently under way, and the staff has not stated a position beyond identifying concerns.

Accident scenarios for existing plants reach a point when reliance must be placed on offsite materials. We expect that the staff will need to be satisfied that the AP600 design can be brought to a stable condition using onsite equipment, and that any additional needed resources are reasonably available.

VII. Containment Performance

The staff intends to use both deterministic and probabilistic containment performance goals in reviewing the AP600. This is consistent with the Commission direction given in the July 21, 1993 Staff Requirements Memorandum related to SECY-93-087. We believe that the staff position is appropriate.

VIII. External Reactor Vessel Cooling

Westinghouse proposes a severe accident mitigation strategy for the AP600 that includes the ability to flood the cavity under the reactor to a level that is effective in cooling the lower reactor vessel shell and preventing reactor vessel melt-through following core melt. The staff stated that this would be a desirable feature if the technical issues can be resolved. The staff is pursuing those issues with Westinghouse. We believe that the staff is following an appropriate path, but we will closely follow the resolution of the technical issues.

IX. Passive Hydrogen Control Measures

The proposed AP600 design includes unpowered catalytic recombiners to control hydrogen generated in a design-basis accident (DBA). This is consistent with the overall concept of controlling design-basis accidents with passive measures. (The plan is to use igniters to control severe accident hydrogen.) There are technical questions involving the qualification and effectiveness of catalytic recombiners in an accident environment. The staff proposes to approve the use of passive recombiners contingent on the resolution of these issues. We believe that the staff position is appropriate.

X. DBA and Long-Term Severe Accident Radiological Consequences

While the passive nature of the AP600 safety features is very attractive, the design has some downside characteristics. Post-accident pressure in the containment will remain positive longer than a plant designed with active cooling. Further, following severe accidents, the removal of radioactive species from the containment atmosphere is expected to be less efficient with passive means than it would be using active sprays or filters. Thus, there is the potential for radioactive leakage for an extended period, compared to that of the existing plants. The staff believes that this situation calls for consideration of additional means, such as a nonsafety-grade containment spray, to reduce containment pressure and suspended radionuclides following a severe accident. The staff has asked Westinghouse to reconsider its proposed position in this regard.

In addition, Westinghouse proposes a source term somewhat different from what the staff would use with respect to both timing and release fractions. The staff indicates that the technical differences here would not be of much concern if the staff can be satisfied that there would be an active system available to reduce the containment leakage potential.

We believe that the issues associated with the potential for radioactive leakage and the source term should be treated separately. We believe that the staff position on the source term is appropriate. The radioactive leakage from the proposed containment design, however, should be considered with respect to public risk and the safety goals.

In the course of this review, it has occurred to us that the certification of advanced light-water reactors provides an important opportunity to continue the evolution toward performance-based regulation. Current plans, unfortunately, do not take complete advantage of this opportunity, perhaps because of schedule constraints. The debate over the procedure to impose unquantified levels of conservatism on analyses of leak-before-break for small-diameter piping reflects a continuation of past practice. The aspirations of both the industry and the NRC would be better served by a performance-based criterion. Similarly, arguments on the time frame for analyses of radionuclide concentrations in containment would be unnecessary if a performance-based criterion were derived. In general, such performance-based criteria would be more consistent with the state-of-the-art engineering being employed in the design of advanced light-water reactors than the continued use of traditional criteria developed in the past when there was a poorer understanding of safety-related processes and phenomena.

Dr. Dana A. Powers did not participate in the Committee's deliberations regarding the severe accident source term. Dr. Thomas S. Kress did not participate in the Committee's deliberations regarding external reactor vessel cooling.

Sincerely,

T. S. Kress
Chairman

<u>References</u>:

1. Memorandum dated May 15, 1995, from J. Taylor, NRC Executive Director for Operations, to the Commissioners, Subject: Advance Information Copy of Forthcoming Commission Paper - Staff Positions on Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design
2. SECY-93-087 dated April 2, 1993, from J. Taylor, NRC Executive Director for Operations, to the Commissioners, Subject: Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs
3. SRM dated July 21, 1993, from S. Chilk, Secretary of the Commission, to J. Taylor, NRC Executive Director for Operations, Subject: SECY-93-087 - Policy, Technical, and

Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs

June 16, 1995

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

Dear Chairman Selin:

SUBJECT: PROPOSED FINAL POLICY STATEMENT ON THE USE OF PROBABILISTIC RISK ASSESSMENT METHODS IN NUCLEAR REGULATORY ACTIVITIES

During the 422nd meeting of the Advisory Committee on Reactor Safeguards, June 8-10, 1995, we reviewed the proposed final Policy Statement on the Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities. We had the benefit of presentations by the NRC staff concerning the resolution of public comments as well as comments we made on a draft version of the Policy Statement. We also had the benefit of presentations by representatives of the Nuclear Energy Institute concerning a draft PSA Applications Guide. Finally, we had the benefit of the referenced documents.

We support a policy statement that encourages the use of probabilistic risk assessment (PRA) methods in nuclear regulatory activities. A policy statement that extends the use of such methods beyond the regulation of nuclear power reactors into other areas within the jurisdiction of the NRC provides a welcome opportunity to improve both the efficiency and the effectiveness of the body of the NRC regulations. Revisions made to the Policy Statement accommodate comments we made on an earlier draft. We feel it useful to issue a policy statement to update positions adopted in the past by the NRC concerning the use of PRA.

We are interested in the challenges that will have to be met to implement the Policy Statement. Technically defensible, risk-based regulatory activities will require the availability of PRAs that are adequately complete and of acceptable quality. Uncertainties in the results of these risk assessments will have to be characterized adequately. The staff indicated that it is aware of these needs. We look forward to hearing more about staff efforts to define standards for PRAs and strategies that will be adopted to audit and to review PRAs submitted by licensees.

The staff is now considering the decision criteria that will be used in conjunction with the application of PRAs. The staff has stated that it feels inhibited from using the NRC safety goals in decisions concerning specific plants. We encourage the use of technically defensible PRA methods for risk management of individual plants consistent with the NRC safety goals. We note that, in such applications, these goals should not be treated as safety criteria. We believe that plant-specific risk management is an important subject which we plan to pursue. We will report on our findings in the future.

The widespread use of PRA methods within the NRC will necessitate a cultural change within the agency. The staff will have to be receptive to different approaches to given issues by different licensees. Training for the staff may need to be on more than PRA applications and methods. For instance, training in formal decision analysis methods may also assist the needed change in culture at the NRC. We are interested in the full scope of the training program in PRA being developed for the NRC staff. We plan to review this training program and the PRA research program that NRC supports.

The Policy Statement calls for the consideration of the use of PRA methods in areas where these methods have not heretofore been extensively used. Consequently, the methods for these new applications are not as well developed as they are for application to nuclear power plants. The NRC may need to support an expanded research effort in the development of PRA methods for application in these new areas.

Sincerely,

T. S. Kress
Chairman

<u>References</u>:

1. SECY-95-126 dated May 18, 1995, from James M. Taylor, Executive Director for Operations, NRC, for the Commissioners, Subject: Final Policy Statement on the Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities (Draft Predecisional)
2. ACRS Report dated May 11, 1994, from T. S. Kress, Chairman, ACRS, to James M. Taylor, Executive Director for Operations, NRC, Subject: Draft Policy Statement on the Use of Probabilistic Risk Assessment Methods in Reactor Regulatory Activities

3. Letter dated January 17, 1995, from William H. Rasin, Nuclear Energy Institute, to Ashok C. Thadani, Office of Nuclear Reactor Regulation, NRC, transmitting final draft of PSA Applications Guide
4. ACRS Report dated May 13, 1987, from William Kerr, Chairman, ACRS, to The Honorable Lando W. Zech, Chairman, NRC, Subject: ACRS Comments On An Implementation Plan For The Safety Goal Policy

June 16, 1995

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

Dear Mr. Taylor:

SUBJECT: PROPOSED FINAL RULE AND REGULATORY GUIDE FOR FRACTURE TOUGHNESS REQUIREMENTS FOR LIGHT WATER REACTOR PRESSURE VESSELS

During the 422nd meeting of the Advisory Committee on Reactor Safeguards, June 8-10, 1995, we discussed the subject rule and regulatory guide. We had the benefit of discussions with representatives of the NRC staff. We also had the benefit of the documents referenced.

The need for the timely development of guidance and requirements for the thermal annealing of reactor pressure vessels (RPVs) became apparent during consideration of the fracture toughness of the RPV at the Yankee Nuclear Power Station. The recent review of the data for the Palisades RPV suggests that variability in the composition of welds and, hence, the uncertainty in the estimation of pressurized thermal shock reference temperature (RT_{PTS}) is greater than previously considered. The result of this review adds greater weight to the need for appropriate regulatory guidance on thermal annealing.

We reviewed a draft version of the rule and the regulatory guide for fracture toughness requirements during our September 1993 meeting. A number of changes have been made in the rule and regulatory guide as a result of public comments. These changes do not affect our technical assessment that the rule and regulatory guide should prove useful to the licensees and the NRC staff, and we believe they should be issued. We also support the proposed changes to Appendix H of 10 CFR Part 50 and the pressurized thermal shock rule (10 CFR 50.61).

We have no objection to the changes in Appendix G that are intended to clarify and restructure the current requirements. We believe, however, that the prohibition against using nuclear heat to conduct ASME Section XI pressure and leak tests of boiling water reactor pressure vessels merits re-examination. It is not at all apparent that this prohibition can be justified in terms of risk. Indeed, there is reason to believe that there could be a reduction in risk in view of the increased requirements for containment and emergency core cooling for critical reactors. We recommend that a probabilistic assessment be performed. Since the practice of using nuclear heat is currently prohibited, an explicit statement in Appendix G is unnecessary and would restrict future action based upon the results of the probabilistic assessment. However, we do not wish this reassessment to delay publication of the thermal annealing rule, the amendment to Appendix H, or the amended pressurized thermal shock rule.

Sincerely,

T. S. Kress
Chairman

References:

1. Letter dated September 20, 1993, from J. Wilkins, Jr., Chairman, ACRS, to J. Taylor, Executive Director for Operations, NRC, Subject: Proposed Rule and Regulatory Guide for Fracture Toughness Requirements
2. Memorandum dated May 23, 1995, from L. Shao, Director, Division of Engineering Technology, RES, to J. Larkins, Executive Director, ACRS, Subject: Request for ACRS Review of Final Rule and Regulatory Guide for Fracture Toughness Requirements for Light Water Reactor Pressure Vessels, with the following attachments:

   - Amendment to 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events"

   - Amendment to 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements"

   - Amendment to 10 CFR Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements"

   - Final Rule (10 CFR 50.66), "Requirements for Thermal Annealing of the Reactor Pressure Vessel"

   - Proposed Regulatory Guide 1.XXXX, "Format and Content of Report for Thermal Annealing of Reactor Pressure Vessels"

June 16, 1995

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

Dear Chairman Selin:

SUBJECT: PROPOSED FINAL REVISIONS TO APPENDIX J OF 10 CFR PART 50, "PRIMARY REACTOR CONTAINMENT LEAKAGE TESTING FOR WATER-COOLED POWER REACTORS"

During the 422nd meeting of the Advisory Committee on Reactor Safeguards, June 8-10, 1995, we considered the changes made to the proposed final revisions to Appendix J in response to public comments. These changes did not alter our views expressed in the report dated September 19, 1994. We find no need to meet again with the staff on this subject and stand by our previously expressed position.

Sincerely,

T. S. Kress
Chairman

References:

1. Memorandum dated June 6, 1995, from Joseph A. Murphy, Executive Assistant to the Director, RES, to John T. Larkins, Executive Director, ACRS, Subject: ACRS Review of Final Amendment to Appendix J of 10 CFR Part 50 (Draft Predecisional Attachment)

2. ACRS Report dated September 19, 1994, from T. S. Kress, Chairman, ACRS, to The Honorable Ivan Selin, Chairman, NRC, Subject: Proposed Revisions to Appendix J of 10 CFR Part 50, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors"