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

ACRSR-1632

PDR

February 27, 1996

The Honorable Albert Gore, Jr.
President of the United States
Senate
Washington, D.C. 20510

Dear Mr. President:

In accordance with the requirements of Section 29 of the Atomic Energy Act of 1954, as amended by Section 5 of Public Law 95-209, the Advisory Committee on Reactor Safeguards reports to the Congress each year on the Safety Research Program of the Nuclear Regulatory Commission (NRC).

In 1995 we reviewed selected NRC research programs and related activities. Much of the research sponsored by the NRC is directed toward improving the current licensing process and providing the technical bases needed to develop risk-informed regulation consistent with the objectives of the National Performance Review. Enclosed are copies of the reports that we have provided to the NRC during the past year that relate to the research program or have suggestions for needed research.

Sincerely,

T. S. Kress
Chairman

Enclosures:
1. Report from T. S. Kress, ACRS Chairman, to James M. Taylor, NRC Executive Director for Operations, Subject: Reactor Water Cleanup System Line Break for Operating BWRs, February 15, 1995
2. Report from T. S. Kress, ACRS Chairman, to James M. Taylor, NRC Executive Director for Operations, Subject: Proposed Final Amendment to 10 CFR 50.55a to Incorporate by Reference Subsections IWE and IWL, Section XI, Division 1, of the ASME Boiler and Pressure Vessel Code, February 17, 1995
3. Report from T. S. Kress, ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: Proposed Rulemaking - Revision to 10 CFR Parts 2, 50, and 51 Related to Decommissioning of Nuclear Power Reactors, March 17, 1995

9603050296 960227
PDR ACRS
R-1632 PDR

0038

RS01
1/1

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

February 27, 1996

The Honorable Newt Gingrich
Speaker of the United States
House of Representatives
Washington, D.C. 20515

Dear Mr. Speaker:

In accordance with the requirements of Section 29 of the Atomic Energy Act of 1954, as amended by Section 5 of Public Law 95-209, the Advisory Committee on Reactor Safeguards reports to the Congress each year on the Safety Research Program of the Nuclear Regulatory Commission (NRC).

In 1995 we reviewed selected NRC research programs and related activities. Much of the research sponsored by the NRC is directed toward improving the current licensing process and providing the technical bases needed to develop risk-informed regulation consistent with the objectives of the National Performance Review. Enclosed are copies of the reports that we have provided to the NRC during the past year that relate to the research program or have suggestions for needed research.

Sincerely,

T. S. Kress
Chairman

Enclosures:
1. Report from T. S. Kress, ACRS Chairman, to James M. Taylor, NRC Executive Director for Operations, Subject: Reactor Water Cleanup System Line Break for Operating BWRs, February 15, 1995
2. Report from T. S. Kress, ACRS Chairman, to James M. Taylor, NRC Executive Director for Operations, Subject: Proposed Final Amendment to 10 CFR 50.55a to Incorporate by Reference Subsections IWE and IWL, Section XI, Division 1, of the ASME Boiler and Pressure Vessel Code, February 17, 1995
3. Report from T. S. Kress, ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: Proposed Rulemaking - Revision to 10 CFR Parts 2, 50, and 51 Related to Decommissioning of Nuclear Power Reactors, March 17, 1995

4. Report from T. S. Kress, ACRS Chairman, to James M. Taylor, NRC Executive Director for Operations, Subject: NRC Test and Analysis Program in Support of AP600 Advanced Light Water Passive Plant Design Review, April 12, 1995
5. Report from T. S. Kress, ACRS Chairman, to James M. Taylor, NRC Executive Director for Operations, Subject: Proposed Final Generic Letter 95-XX, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes," May 15, 1995
6. Report from T. S. Kress, ACRS Chairman, to James M. Taylor, NRC Executive Director for Operations, Subject: Review of Best-Estimate Models for Evaluation of Emergency Core Cooling System Performance, May 17, 1995
7. Report from T. S. Kress, ACRS Chairman, to James M. Taylor, NRC Executive Director for Operations, Subject: Proposed Commission Paper on Staff Positions on Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design, June 15, 1995
8. Report from T. S. Kress, ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: Proposed Final Policy Statement on the Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities, June 16, 1995
9. Report from T. S. Kress, ACRS Chairman, to James M. Taylor, NRC Executive Director for Operations, Subject: Health Effects Valuation, July 20, 1995
10. Report from T. S. Kress, ACRS Chairman, to James M. Taylor, NRC Executive Director for Operations, Subject: The Nuclear Energy Institute Petition for Rulemaking to Amend 10 CFR 50.48, "Fire Protection," September 15, 1995
11. Report from T. S. Kress, ACRS Chairman, to Shirley A. Jackson, NRC Chairman, Subject: Development of Improved Nondestructive Examination (NDE) Techniques, September 15, 1995
12. Report from T. S. Kress, ACRS Chairman, to Shirley A. Jackson, NRC Chairman, Subject: National Academy of Sciences/National Research Council Study on "Digital Instrumentation and Control Systems in Nuclear Power Plants, Safety and Reliability Issues" - Phase 1, October 13, 1995
13. Report from T. S. Kress, ACRS Chairman, to Shirley A. Jackson, NRC Chairman, Subject: Fatigue Action Plan, October 16, 1995
14. Report from T. S. Kress, ACRS Chairman, to James M. Taylor, NRC Executive Director for Operations, Subject: NUREG-0700, Revision 1, "Human-System Interface Design Review Guideline," November 13, 1995
15. Report from T. S. Kress, ACRS Chairman, to James M. Taylor, NRC Executive Director for Operations, Subject: Proposed Final Regulatory Guide 1.164, "Time Response Design Criteria for Safety-Related Operator Actions," to Resolve Generic Safety Issue B-17, November 14, 1995

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

February 15, 1995

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

Dear Mr. Taylor:

SUBJECT: REACTOR WATER CLEANUP SYSTEM LINE BREAK FOR OPERATING BWRS

During the 418th meeting of the Advisory Committee on Reactor Safeguards, February 9-10, 1995, we held discussions with representatives of the NRC staff concerning Issue 3 [Reactor Water Cleanup (RWCU) Systems Safety] from our letter to you dated July 13, 1994 (Reference 1). In our letter, we pointed out that an added RWCU isolation valve inside primary containment provides long-term post-accident isolation of the ABWR if the primary containment isolation valves fail to close fully under blowdown conditions resulting from an RWCU line break outside of primary containment. We suggested that operating plants may not have a similar capability and recommended that this issue be investigated.

In your September 9, 1994 response (Reference 2), you stated that the staff will perform a study to determine whether the environmental conditions in secondary containment resulting from an RWCU line break would create an environment bounded by the current analyses for operating plants. We discussed this response with the NRC staff members. They assured us that the environmental conditions would include those associated with the postulated event described below.

> For this event, a pipe break is postulated in the safety or non-safety portion of the RWCU system outside of primary containment. A blowdown of reactor coolant and steam to the break occurs until the break is isolated by containment isolation valves. If these valves are unable to close completely due to the severity of simultaneous mechanical and electrical demands on both valves under blowdown flow conditions, the reactor will continue to discharge a portion of its coolant and steam inventory to the break indefinitely.
>
> It is likely that several remotely operated relief valves on the reactor steam lines will be opened to divert a

9503020082 3pp.

ENCLOSURE 1

portion of the steam directly to the suppression pool. However, for a typical BWR-4 (and perhaps for many other BWRs) these relief valves will close at about 50 psig even if they are externally actuated to open. The valves will not reopen until the reactor repressurizes to about 85 psig.

If the ECCS pumps are operating, the water flowing into the reactor vessel may increase the vessel pressure sufficiently to lift and hold open the remotely operated relief valves. This should ensure adequate core cooling while the pumps are running, but a significant portion of the ECCS flow will be diverted to the unisolated break thereby depleting the water inventory needed to ensure proper pump operation during long-term core cooling. In addition, the diverted water will be released inside of secondary containment where it can gravitate to the lowest level where the ECCS pumps and drivers are located. The resulting water cascading and flooding may jeopardize the continued availability of the ECCS pumps and equipment during long-term core cooling.

If adequate ECCS flow is not maintained, core uncovery to below the level of the jet pump throat (2/3 core level) is a certainty. (The reactor coolant loss will be greater if the reactor vessel bottom drain line is open and cannot be closed.) If the ECCS pumps are not operating, the relief valves will cycle in the 50-85 psig range. Still, a portion of the reactor coolant will be diverted to the break. Eventually, the fuel decay heat will be insufficient to repressurize the reactor to 85 psig. Thereafter, the relief valves will remain closed and any ECCS flow and resulting steam will be directed to the break.

Various corrective actions or features might be considered to mitigate this event, but most have shortcomings. For example, one could provide remotely operated relief valves which can be kept open during the event. Since the relief valves exhaust to the suppression pool, the reactor pressure must be sufficient to overcome the drywell pressure and the pressure equivalent of the relief valve sparger submersion depth. Although dependent on the piping arrangement to the break, the reactor pressure may be sufficient to direct most ECCS water and steam from the core to the break. Provisions for relieving directly to the containment atmosphere could overcome this problem only if the containment is maintained at essentially the same pressure as at the break location and if the piping arrangement to the break is not conducive to siphoning. Opening the main steam lines to a functional main condenser (if operating at partial vacuum) might be a solution if it were possible to arrange when subject to the human

and equipment limitations created by the break and harsh environment in primary and secondary containment. Other solutions may be proposed.

We believe that the primary containment isolation valves for the RWCU system must be able to perform their safety function while subjected to the conditions present when the valves are required to operate. We agree that the ability of these valves to perform their design function was considered in the resolution of Generic Issue 87, "HPCI Steam Line Break Without Isolation." We also agree that the implementation of Generic Letter No. 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," should improve the likelihood of proper valve functioning under design-basis conditions. We are concerned, however, that sufficient test data under actual blowdown flow conditions and realistic geometries are not available to validate the valve reliability used in current probabilistic risk assessments.

We are concerned that the risk associated with an RWCU pipe break outside of primary containment has been underestimated and that a need may exist for additional isolation capability in the RWCU line inside of primary containment. We look forward to seeing the results of the current investigations. We recommend that similar studies be undertaken of the risk significance of failure to isolate high energy line breaks outside primary containment in the High Pressure Coolant Injection and Reactor Core Isolation Cooling Systems.

Sincerely,

T. S. Kress
Chairman

References:

1. Letter dated July 13, 1994, from T. S. Kress, ACRS Chairman, to James M. Taylor, NRC Executive Director for Operations, Subject: Some Areas for Potential Staff Consideration for Operating Nuclear Power Plants and the Review of Future Plant Designs Resulting from the ACRS Review of the Evolutionary Light Water Reactors
2. Memorandum dated September 9, 1994, from James M. Taylor, NRC Executive Director for Operations, to T. S. Kress, ACRS Chairman, Subject: Some Areas for Potential Staff Consideration for Operating Nuclear Power Plants and the Review of Future Plant Designs Resulting from the ACRS Review of the Evolutionary Light Water Reactors

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

February 17, 1995

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

Dear Mr. Taylor:

SUBJECT: PROPOSED FINAL AMENDMENT TO 10 CFR 50.55a TO INCORPORATE BY REFERENCE SUBSECTIONS IWE AND IWL, SECTION XI, DIVISION 1, OF THE ASME BOILER AND PRESSURE VESSEL CODE

During the 418th meeting of the Advisory Committee on Reactor Safeguards, February 9-10, 1995, we discussed the subject final amendment. At this meeting, we had discussions with representatives of the NRC staff and the Nuclear Energy Institute (NEI). We also had the benefit of the document referenced.

This proposed final amendment incorporates by reference the 1992 Edition with the 1992 Addenda of Subsection IWE (Requirements for Class MC and Metallic Liners of Class CC Components of Light-Water Cooled Power Plants) and Subsection IWL (Requirements for Class CC Concrete Components of Light-Water Cooled Power Plants), Section XI, Division 1, of the ASME Boiler and Pressure Vessel Code with specified modifications and a limitation. It also expedites the schedule for performing the containment examinations. We concur with this staff position.

A number of utilities and NEI, which have commented on a draft version of this amendment, argue that it is overly prescriptive and contrary to the trend towards performance-based regulation. However, a suitable "metric," which could be used as the basis for a performance-based inspection for the assurance of the structural integrity of the containment, seems difficult to identify. Risk-based inspection appears to be a more promising approach to rationalizing in-service inspection of passive structural components. The Office of Nuclear Regulatory Research is actively pursuing this approach, and we hope to see risk-based concepts

ENCLOSURE 2

9503020088 2pp.

being used to develop requirements for in-service inspections in the not-too-distant future.

Sincerely,

T. S. Kress
Chairman

Reference:
Memorandum dated December 12, 1994, from E. Beckjord, Director, Office of Nuclear Regulatory Research, to J. Larkins, Executive Director, ACRS, Subject: Final Amendment to 10 CFR 50.55a to Incorporate by Reference Subsection IWE and Subsection IWL, Section XI, Division 1, of the ASME Boiler and Pressure Vessel Code

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

March 17, 1995

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

Dear Chairman Selin:

SUBJECT: PROPOSED RULEMAKING - REVISION TO 10 CFR PARTS 2, 50, AND 51 RELATED TO DECOMMISSIONING OF NUCLEAR POWER REACTORS

During the 419th meeting of the Advisory Committee on Reactor Safeguards, March 9-10, 1995, we reviewed the proposed rule on decommissioning of nuclear power reactors. During our review, we had discussions with representatives of the NRC staff and the Nuclear Energy Institute. We had the benefit of the document referenced.

The proposed revision to the decommissioning rule appears to allow significant flexibility for different possible circumstances under which a nuclear plant may cease operation and transition into the decommissioning mode. The proposed revision to the rule reduces unnecessary burdens on both the licensees and NRC staff.

We believe that the proposed rule should be issued for public comment. We are concerned, however, that the proposed rule has not been founded on a risk basis. Realistic risk analyses for decommissioning nuclear power reactors have not been done. Consequently, there is no clear relationship between the requirements being retained in the revised rule and the realistic risks to the public health and safety and the environment posed by decommissioning. The revised rule may still impose unnecessary burdens on licensees and may make excessive demands on NRC resources. We hope that steps can be taken in the near future to establish a risk basis for reformulating 10 CFR Parts 2, 50, and 51. We believe this is an issue on which comment from the industry and the public should be sought.

Sincerely,

T. S. Kress
Chairman

9503290282 2pp.

ENCLOSURE 3

Reference:
Memorandum dated January 27, 1995, from Bill Morris, Director, Division of Regulatory Applications, RES, to John Larkins, Executive Director ACRS, forwarding Proposed Rule to Amend 10 CFR Parts 2, 50, and 51

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

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

9505020173 3pp.

ENCLOSURE 4

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

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

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

May 15, 1995

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

Dear Mr. Taylor:

SUBJECT: PROPOSED FINAL GENERIC LETTER 95-XX, "VOLTAGE-BASED REPAIR CRITERIA FOR WESTINGHOUSE STEAM GENERATOR TUBES"

During the 421st meeting of the Advisory Committee on Reactor Safeguards, May 4-6, 1995, we discussed the subject generic letter. During this meeting, we had the benefit of discussions with representatives of the NRC staff, the Nuclear Energy Institute, and the Southern Nuclear Operating Company. We also had the benefit of the documents referenced.

We provided comments on a draft version of the generic letter in our report dated September 12, 1994. A number of changes have been made in the generic letter as a result of public comments. These changes do not affect our technical assessment that the generic letter provides an acceptable approach to ensure the integrity of tubing subject to axially oriented outside diameter stress corrosion cracking in Westinghouse steam generators with drilled-hole support plates.

In our September 12, 1994 report, we noted that the database for the present empirical correlations of burst pressure, leakage, and bobbin coil voltage appears to be only marginally adequate. Because of this, we believe the staff decision to retain the conservative lower voltage limits of 2 volts for 7/8-inch diameter tubing and 1 volt for 3/4-inch diameter tubing until more experience is gained with the application of the criteria is prudent and appropriate.

In our previous report, we noted that the concern raised in the differing professional opinion on the calculation of the radiological releases during a main steamline break appeared to warrant further consideration. This issue has not yet been resolved, but we believe that timely implementation of the generic letter should proceed to prevent unnecessary tube repairs and reduce staff resources associated with plant-specific reviews. However, the radiological release issue should be addressed in the proposed rule on steam generator tube maintenance and surveillance.

ENCLOSURE 5

9505260201 2 1p.

Dr. William Shack did not participate in the Committee's deliberations regarding this matter.

Sincerely,

T. S. Kress
Chairman

References:

1. Memorandum dated April 6, 1995, from Brian Sheron, Director, Division of Engineering, NRR, to John Larkins, Executive Director, ACRS, Subject: ACRS Review of Generic Letter (GL) 95-XX, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes"
2. ACRS Report dated September 12, 1994, from T. S. Kress, Chairman, ACRS, to Ivan Selin, Chairman, NRC, Subject: Proposed Generic Letter 94-XX, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes"

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

May 17, 1995

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

Dear Mr. Taylor:

SUBJECT: REVIEW OF BEST-ESTIMATE MODELS FOR EVALUATION OF EMERGENCY CORE COOLING SYSTEM PERFORMANCE

During the 421st meeting of the Advisory Committee on Reactor Safeguards, May 4-6, 1995, we discussed the methodology being applied by NRR for reviewing the acceptability of best-estimate calculations of emergency core cooling system (ECCS) performance in accordance with the revisions made to 10 CFR 50.46 (ECCS Rule). Our Subcommittee on Thermal Hydraulic Phenomena held a meeting on May 2, 1995, to discuss this matter. During these meetings, we had the benefit of discussions with representatives of NRR and the Westinghouse Electric Corporation. We also had the benefit of the documents referenced.

A historical impediment to the use of best-estimate predictions of plant behavior following a large-break LOCA was the lack of a method for determining the accuracy of the predicted peak cladding temperature. In a September 16, 1986 report, the ACRS made the following comment:

> "The acceptability of realistic evaluation models rests on the development of a satisfactory methodology for determination of the code overall uncertainty. . . . We recommend that the methodology used to evaluate uncertainty be subjected to peer review."

This was done and the ACRS reviewed and endorsed the resulting Code Scaling, Applicability, and Uncertainty (CSAU) evaluation methodology. It is our view that the CSAU methodology provides a well-structured, traceable, and practical technical basis for quantifying best-estimate code uncertainty. It was the development and demonstration of the CSAU methodology that allowed the successful promulgation of the revision to the ECCS Rule.

Westinghouse Electric Corporation is presenting an alternative approach to the CSAU methodology for determining the uncertainty in

9505260204 3pp.

ENCLOSURE 6

its best-estimate computer code predictions for both existing plants and the AP600 passive plant design. This best-estimate code is intended to meet the requirement of the ECCS Rule that to a "high level of probability," the ECCS criteria will not be exceeded. Although the ECCS Rule allows alternative approaches, none has been reviewed to date nor have review criteria been developed. If Westinghouse persists in following its present path, it is unclear if the intent of 10 CFR 50.46 will be met. Based on the staff presentations, it appears that adoption of the alternative approach would require a weakening of the acceptance criterion for evaluating uncertainty. We believe the staff should be able to confirm that the Westinghouse uncertainty evaluation conforms to the applicable requirements of 10 CFR 50.46 Paragraph (a)(1)(i) in terms of both high probability and high confidence.

During our meeting, we learned that at least two more applicants are requesting approval of best-estimate computer codes. We do not know how they plan to address the nonexceedance requirement of 10 CFR 50.46. A clear statement is needed from the staff as to what constitutes an acceptable demonstration that the ECCS nonexceedance criterion has been met. We would like to see such a statement before the staff begins its review of these other best-estimate codes.

Several aspects of the current review process that were discussed during our meeting should be noted. The review of the Westinghouse best-estimate code has been under way since 1992. We were told that during this period, there has been no formal documentation of this review. Key elements of the alternative approach proposed by Westinghouse for uncertainty have not been addressed. The material submitted by Westinghouse in support of its best-estimate code application is confusing and difficult to follow.

The staff waits for Westinghouse to present its arguments and then reacts as best it can, using some of the provisions of Regulatory Guide 1.157 to guide the review. This reactive approach is a risky procedure for both Westinghouse and the staff. Furthermore, it is much more resource intensive to both because of the iterative nature of "wait-and-see," followed by rounds of questions and answers. This process is time consuming, unstructured, and difficult to trace.

We recommend prompt attention to these matters.

Sincerely,

T. S. Kress
Chairman

References:

1. 10 CFR 50.46(a), as amended through August 31, 1992, Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors

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

3. Westinghouse Electric Corporation Report Addressing Compliance of the Westinghouse Best-Estimate LBLOCA Code and Methodology described in WCAP-12945-P with NRC Regulatory Positions Described in Regulatory Guide 1.157 (**Westinghouse Proprietary**), transmitted by telecopy from Westinghouse Electric Corporation dated March 31, 1995

4. Table 2.1.2-1, Comparison of Regulatory Guide 1.157 Requirements and Westinghouse's Best-Estimate Large-Break LOCA Model (Draft), transmitted by telecopy from INEL dated March 23, 1995

5. Westinghouse Response to Requests for Additional Information on WCAP-12945-P, Volume 5, COBRA/TRAC Code Qualification Document, transmitted by telecopy from INEL dated April 12, 1995, [Westinghouse Proprietary]

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

7. Letter dated April 24, 1995, from L. W. Ward, INEL, to F. Orr, Office of Nuclear Reactor Regulation, NRC, transmitting Draft Westinghouse Report, "Review and Evaluation, Westinghouse Code Qualification for Best Estimate LOCA Analysis," dated April 24, 1995

8. **ACRS Report** dated September 16, 1986, from D. A. Ward, **Chairman**, ACRS, to L. Zech, Jr., Chairman, NRC, Subject: ACRS **Comments** on the Proposed Revision to the ECCS Rule in 10 CFR 50.46, "Acceptance Criteria for ECCS for Light Water Nuclear Power Reactors," and Appendix K, "ECCS Evaluation Models"

9. ACRS Report dated July 20, 1988, from W. Kerr, Chairman, ACRS, to V. Stello, Jr., Executive Director for Operations, NRC, Subject: Comments on the Staff's Draft Safety Evaluation of the Westinghouse Topical Report, WCAP-10924, "Westinghouse Large-Break LOCA Best-Estimate Methodology"