ARLEN SPECTER
PENNSYLVANIA

COMMITTEES
JUDICIARY
APPROPRIATIONS
VETERANS' AFFAIRS
INTELLIGENCE

United States Senate

WASHINGTON, DC 20510

February 28, 1991

Mr. Dennis K. Rathbun
Director, Congressional Affairs,
Office of Government and Public Affairs
Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Rathbun:

I am writing on behalf of my constituent, Mr. Michael J. Becker, of Media, Pennsylvania. The information contained herein was provided to me by Mr. Becker.

Mr. Becker is concerned about he Savannah River Plant reactors. Enclosed are materials provided to me by Mr. Becker.

Please accord this case all due consideration. I would appreciate it if the Nuclear Regulatory Commission would respond to Mr. Becker directly and forward a copy of the response to the attention of Susan Becker of my staff.

Thank you for your attention to this matter.

Sincerely,

Arlen Specter

Arlen Specter

AS:srb
Enclosure

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-> NRC

Dear Sir

I urge you to prevent the restart of any of the Savannah River Plant reactors until the D.O.E. is in full compliance with all the applicable environmental laws and the Office of Technology Assessment completes a report on actual tritium requirements.

Michael J Becker

[redacted]

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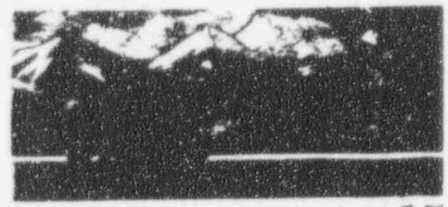

America the Beautiful USA 15

Senator Arlen Specter
342 Senate Office Bldg.
Washington, D.C. 20510

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

January 24, 1995

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 (ACRS) has reported to the Congress each year on the Safety Research Program of the Nuclear Regulatory Commission (NRC). In our December 18, 1986, letter to the Congress, we proposed to provide reports on specific issues rather than one all-inclusive report, as we had provided before 1986.

In 1994 we reviewed selected NRC research programs and related activities. Much of this work was directed toward the understanding of the conservatisms used in the NRC licensing process. Enclosed are copies of the reports that we have provided to the NRC during the past year on these matters. We expect to continue to review various elements of the NRC Safety Research Program and provide reports to the Commission as warranted.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

Enclosures:
1. Report from J. Ernest Wilkins, Jr., ACRS Chairman, to Ivan Selin, U.S. NRC Chairman, Subject: Draft Commission Paper on Source Term Related Technical and Licensing Issues Pertaining to Evolutionary and Passive Light Water Reactor Designs, March 15, 1994
2. Report from T. S. Kress, ACRS Chairman, to James M. Taylor, Executive Director for Operations, Subject: Draft Policy Statement on the Use of Probabilistic Risk Assessment Methods in Reactor Regulatory Activities, May 11, 1994
3. Report from T. S. Kress, ACRS Chairman, to James M. Taylor, Executive Director for Operations, Subject: Proposed Rule for Shutdown and Low-Power Operations, May 13, 1994

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4. Report from T. S. Kress, ACRS Chairman, to Ivan Selin, U.S. NRC Chairman, Subject: Thermo-Lag Fire Barriers, June 14, 1994
5. Report from T. S. Kress, ACRS Chairman, to Ivan Selin, U.S. NRC Chairman, Subject: Emergency Planning Zones, Protective Action Guidelines, and the New Source Terms, July 13, 1994
6. Report from T. S. Kress, ACRS Chairman, to James M. Taylor, 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, July 13, 1994
7. Report from T. S. Kress, ACRS Chairman, to Ivan Selin, U.S. NRC Chairman, Subject: Proposed National Academy of Sciences/National Research Council Study and Workshop on Digital Instrumentation and Control Systems, July 14, 1994
8. Report from T. S. Kress, ACRS Chairman, to Ivan Selin, U.S. NRC Chairman, Subject: Proposed Generic Letter 94-XX, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes," September 12, 1994
9. Report from T. S. Kress, ACRS Chairman, to James M. Taylor, Executive Director for Operations, Subject: Proposed Generic Letter on the Use of NUMARC/EPRI REPORT TR-102348, "Guideline on Licensing Digital Upgrades," September 14, 1994
10. Report from T. S. Kress, ACRS Chairman, to Ivan Selin, U.S. NRC Chairman, Subject: Revised Regulatory Analysis Guidelines, September 14, 1994
11. Report from T. S. Kress, ACRS Chairman, to Ivan Selin, U.S. NRC Chairman, Subject: Proposed Revisions to Appendix J to 10 CFR Part 50, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," September 19, 1994
12. Report from W. J. Lindblad, ACRS Vice-Chairman, to Ivan Selin, U.S. NRC Chairman, Subject: Proposed Final Version of NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," September 20, 1994
13. Report from T. S. Kress, ACRS Chairman, to Ivan Selin, U.S. NRC Chairman, Subject: Potential for BWR ECCS Strainer Blockage Due to LOCA Generated Debris, October 14, 1994
14. Report 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 Light Water Reactor Passive Plant Design Certification Reviews, November 10, 1994

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

January 24, 1995

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 (ACRS) has reported to the Congress each year on the Safety Research Program of the Nuclear Regulatory Commission (NRC). In our December 18, 1986, letter to the Congress, we proposed to provide reports on specific issues rather than one all-inclusive report, as we had provided before 1986.

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Sincerely,

T. S. Kress

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Chairman

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

March 15, 1994

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

Dear Chairman Selin:

SUBJECT: DRAFT COMMISSION PAPER ON SOURCE TERM RELATED TECHNICAL AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND PASSIVE LIGHT WATER REACTOR DESIGNS

During the 406th and 407th meetings of the Advisory Committee on Reactor Safeguards, February 10-11 and March 10-12, 1994, respectively, we discussed the draft Commission paper on source term related technical and licensing issues pertaining to evolutionary and passive light water reactor (LWR) designs. During these meetings, we had the benefit of discussions with representatives of the NRC staff and industry. We also had the benefit of the documents referenced.

Separate source terms are provided for BWRs and PWRs. The source terms consist of the fraction of the equilibrium core inventory of fission products released into containment, the timing of this release, and the chemical form of the fission product iodine. In the past, such source terms have been specified in Regulatory Guides 1.3 and 1.4 to provide guidance on appropriate values to use in the site suitability analyses that are required by 10 CFR Part 100, and in conjunction with the other design basis accidents (DBAs) in Chapter 15 of the Standard Review Plan. The DBA source terms should not be confused with the plant and sequence specific source terms that are mechanistically derived and used in PRAs and other severe accident analyses. The specifications that are presently in Regulatory Guides 1.3 and 1.4 consist of 100 percent of the noble gases and 25 percent of the iodine (91 percent as elemental iodine, 5 percent as particulate iodine, and 4 percent as organic iodine). For site suitability analyses, these specifications have been used along with a thermal hydraulic specification. These analyses require that a peak containment pressure be calculated for a double-ended break of the largest primary system piping and be applied for 24 hours after which it is to be reduced to half that value.

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The 10 CFR Part 100 specifications of the source term have always been viewed as being somewhat arbitrary, but conservative. The proposed revised source terms are intended to remove some of the arbitrariness of the present values and to make them more realistic. As part of the overall process of decoupling site suitability decisions from reactor design, the revised source term and the dose criteria provisions are to be removed from 10 CFR Part 100 and put into 10 CFR Part 50 where they would apply only to design features. The revised source terms are based on values developed in NUREG-1150 for the "in-vessel" release phase associated with severe accidents.

In the draft Commission paper, the staff describes the proposed revised source terms and proposed uses for reviews and assessments of evolutionary and passive LWR designs. The paper discusses positions taken by the staff on source term issues for evolutionary and passive LWR designs (identified in SECY-90-016 and SECY-93-087). The staff believes these positions will provide a basis for closing these issues with respect to design certification reviews and the EPRI Utility Requirements Documents.

We generally agree with the positions taken by the staff on the issues and agree with the principle that the source terms for DBAs should be made more realistic. Realistic source terms should result in more appropriate designs (e.g., engineered safety features, source term mitigation features, sampling and measurement devices, and containment integrity). We believe the changes can lead to increased coherence in the associated regulations and their application. As in all responses to the accumulation of new knowledge, such proposed changes in the regulations, whether toward enhancement or relaxation, or whether applied to existing plants or to future plants, should be assessed for their overall effect on risk. We also have the following concern about the revised source term specifications.

We think the realistic specification of the thermal hydraulics and production of nonradioactive aerosols associated with the DBAs is as important as the specification of the source term itself. These conditions can strongly influence the behavior of radioactive aerosols in containment. Additional consideration should be given to developing Commission guidance on the thermal hydraulic conditions and nonradioactive aerosol generation to be coupled with the source terms for the various DBAs.

We continue to recommend that the General Design Criteria for containment volume and strength for future ALWRs incorporate the spectrum of severe accident challenges described in our report of

May 17, 1991. The containment should represent a defense-in-depth feature that is not limited to design basis accidents.

Sincerely,

J. Ernest Wilkins, Jr.

J. Ernest Wilkins, Jr.
Chairman

References:

1. Memorandum dated January 6, 1994, from Dennis M. Crutchfield, NRC Office of Nuclear Reactor Regulation, for John T. Larkins, Executive Director, ACRS, Subject: ACRS Review of Commission Paper on Source Term-Related Technical and Licensing Issues Pertaining to Evolutionary and Passive Light-Water-Reactor Designs
2. Memorandum dated February 10, 1994, from James M. Taylor, NRC Executive Director for Operations, for the Commissioners, Subject: Draft Commission Paper, "Source Term Related Technical and Licensing Issues Pertaining to Evolutionary and Passive Light-Water-Reactor Designs"
3. SECY-93-087, Memorandum dated April 2, 1993, from James M. Taylor, Executive Director for Operations, for the Commissioners, Subject: Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs
4. SECY-90-016, Memorandum dated January 12, 1990, from James M. Taylor, Executive Director for Operations, for the Commissioners, Subject: Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements
5. NUREG-1150, Volumes 1 and 2, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power plants," December 1990
6. Report dated May 17, 1991, from David A. Ward, ACRS Chairman, to Kenneth M. Carr, NRC Chairman, Subject: Proposed Criteria to Accommodate Severe Accidents in Containment Design

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

May 11, 1994

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

Dear Mr. Taylor:

SUBJECT: DRAFT POLICY STATEMENT ON THE USE OF PROBABILISTIC RISK ASSESSMENT METHODS IN REACTOR REGULATORY ACTIVITIES

During the 409th meeting of the Advisory Committee on Reactor Safeguards, May 5-7, 1994, we reviewed the current draft Policy Statement on agency usage of probabilistic risk assessment (PRA). We had the benefit of discussions with representatives of the NRC staff. We also had the benefit of the documents referenced.

We are in general agreement with the Policy Statement. It appears to present an appropriate position on the use of PRA in the regulatory process. We are, however, concerned with some aspects of the Policy.

Some provisions of the Policy Statement are crafted in rather weak language. For example, we believe that in Item (2) of Section II, Policy Statement, the word "may" ought to be replaced by "should" to make a commitment to increase the use of PRA to help eliminate unnecessary conservatism associated with current regulatory requirements.

The Policy is very general and does not provide any specific guidance or plan for the expanded use of PRA in regulatory activities. This has apparently been relegated to an "implementation plan" which is referred to in the Policy Statement. We hope that this plan will provide some specific and definitive elements to guide the use of PRA in the regulatory process. We recommend that the implementation plan be submitted for public comment along with the Policy Statement.

The draft Policy Statement seems to draw a distinction between the traditional regulatory process (commonly known as "deterministic") and the PRA approach. This common perception causes some in the regulatory arena to be skeptical of and reluctant to embrace the PRA approach. However, we believe that treating the PRA approach as a distinct and unique method compared to the traditional approach is inappropriate and misleading. We believe that the PRA approach should be considered as an extension and enhancement of traditional regulation rather than a separate and different technology. Certainly, the deterministic approach is replete with implied elements of probability, from the selection of accidents to

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be analyzed (e.g., reactor vessel rupture is too improbable to be considered) to the requirements for emergency core cooling (e.g., safety train redundancy and protection against single failure). The PRA approach enhances traditional approaches by considering risk in a coherent and complete manner, thereby providing a method to quantify the overall level of safety.

We agree that there are uncertainties, limitations, and omissions with the PRA approach. However, we think it is important to understand that these uncertainties are derived from knowledge limitations. These knowledge limitations were not created by PRA, but rather were exposed by it. These limitations existed during the traditional regulatory approach, some were unknown, others only vaguely understood. Attempts were made to accommodate these limitations by imposing prescriptive and what was hoped to be conservative regulatory requirements. The PRA approach has exposed these limitations and has provided a framework to assess their significance and assist in developing a strategy to accommodate them in the regulatory process. We are pleased that these issues are identified in the Policy Statement and that they are being addressed in the implementation plan.

One of the more important shortcomings of PRA use was not identified in the Policy Statement. This is the misuse and misapplication of PRA results stemming from an incomplete and/or flawed analysis. While those in the nuclear regulatory arena have done an excellent job in many instances in applying and using PRA, there have been examples where this has not been the case. Among the more important of these are some of the cost/benefit analyses for backfits. We recognize that these analyses are difficult. We urge the staff to assign high priority in the implementation plan to improving and adding consistency to cost/benefit analyses.

We further believe that the implementation plan needs to address the need for PRA research to help assure that the PRA state-of-the-art is at a level consistent with the intended PRA usage in the agency. We intend to further consider the area of PRA research needs in the near future.

In conclusion, we reiterate our support for the overall thrust of the PRA Policy Statement and the allocation of resources to implement it. We would like to be kept informed of the progress in developing the implementation plan.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

References:

1. Memorandum (Undated) from James M. Taylor, Executive Director for Operations, for The Commissioners, Subject: Draft Policy Statement on the Use of Probabilistic Risk Assessment Methods in Reactor Regulatory Activities, received May 5, 1994 (Predecisional)
2. Memorandum dated April 14, 1994, from Martin J. Virgilio, Office of Nuclear Reactor Regulation, to John T. Larkins, Executive Director, ACRS, Subject: PRA Draft Policy Statement, with Predecisional Enclosure
3. U.S. Nuclear Regulatory Commission, Policy Statement dated January 18, 1979, Subject: NRC Statement on Risk Assessment and The Reactor Safety Study Report (WASH-1400) In Light of the Risk Assessment Review Group Report

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

May 13, 1994

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

Dear Mr. Taylor:

SUBJECT: PROPOSED RULE FOR SHUTDOWN AND LOW-POWER OPERATIONS

During the 409th meeting of the Advisory Committee on Reactor Safeguards, May 5-7, 1994, we reviewed the NRC staff proposed Rule and associated Regulatory Guide pertaining to the conduct of shutdown and low-power operations. During this review, We had the benefit of discussions with representatives of the Office of Nuclear Reactor Regulation and the Office of the General Counsel, the Nuclear Energy Institute (NEI), and the Combustion Engineering Owners Group (CEOG). We have previously commented on the staff program to resolve this issue in our letters dated August 13, 1991, April 9, 1992, and September 15, 1992. We also had the benefit of the documents referenced.

In our September 15, 1992 letter, we commented on three issues that were of concern to us: proposed technical specifications for PWR containment integrity, proposed requirements for fire protection during shutdown, and the adequacy of the staff regulatory analysis. Your letter of October 16, 1992 indicated that the staff was in general agreement with our comments. (At the time of these letters, the staff was planning to utilize a generic letter, instead of rulemaking, to resolve this issue.) In addition, you stated that the staff would provide written responses to five questions raised by the Committee members during an April 1, 1992 Subcommittee meeting. The staff provided this information in a letter dated September 20, 1993, and we concluded that these responses were generally satisfactory.

Our present review has been based on the rulemaking package provided to the Committee to Review Generic Requirements (CRGR) for its review, as supplemented by a revised package containing changes the staff proposes to make in response to the recommendations made by the CRGR. In addition, we considered the views presented by the CEOG in its letter dated April 8, 1994.

The staff now proposes to resolve concerns regarding the conduct of shutdown and low-power operations by rulemaking that would require that licensees (1) plan and control outages in a way that provides

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reasonable assurance that the key safety functions of maintaining the reactor subcritical, removing decay heat, and maintaining reactor coolant system (RCS) inventory will be preserved; (2) establish limiting conditions for operation and surveillance requirements for specific equipment relied on during shutdown and low-power operations; (3) demonstrate, by analysis, that those functions necessary to remove decay heat from the reactor can be maintained during cold shutdown and refueling conditions in the event of a fire in any plant area; (4) install instrumentation for monitoring water level in the RCS of pressurized water reactors during midloop operation.

We believe that improvements are needed in the conduct of shutdown and low-power operations. However, we have concluded that the staff has not made a sufficient case in its regulatory analysis either quantitatively or qualitatively to satisfy the requirements specified in 10 CFR 50.109. Where quantitative support for a backfit decision is not practicable, the use of subjective judgment should be acknowledged and the bases better substantiated than was done in this case.

Many of the staff-proposed improvements appear to have merit; some have already been adopted by the industry; others appear to require additional thought. (The CEOG provided us with data, for the period from 1989 through 1993, that demonstrate a substantial reduction in licensee events occurring during shutdown and involving loss of decay heat removal capability.) We believe that specific requirements of the Rule should continue to be the subject of a dialogue between the staff and NEI and that issuance of the Rule for public comment should be deferred until this dialogue is completed. We also believe that insights from the recently completed PRAs performed under a contract with the Office of Nuclear Regulatory Research should be considered.

Our comments relating to the safety improvements that the staff believes would result from this proposed rulemaking are as follows:

- In the regulatory analysis the staff states that "... a licensee program that (1) fully implements the guidelines in NUMARC 91-06 (Guidelines for Industry Actions to Assess Shutdown Management) and (2) incorporates the features regarding fire protection and instrumentation listed in Table 2.1 would be consistent with the staff assumptions regarding the administrative controls portion of this improvement (Improvement A)."

  NEI believes that the industry initiative, as delineated in the NUMARC 91-06 document, obviates the need for including outage planning and control requirements in this rulemaking. NEI stated during our meeting that all power reactor licensees are implementing these Guidelines. The staff acknowledges that implementation of these Guidelines has been "a significant and constructive step, effects of which have already been realized by many utilities ... in recent outages." We believe that past industry initiatives have proven to be an effective means of resolving safety issues without the need for rulemaking (e.g., Institute of Nuclear Power Operations accreditation of licensee training programs). This leads us to question the need for additional regulation relating to outage planning and control requirements.

- We do not believe that the staff has clearly defined what is expected of licensees relative to fire hazards assessment and associated fire contingency plans, including the bases for such plans. We plan to review the results of the NRC staff reassessment of its fire protection program as discussed in SECY-93-143. Discussion of shutdown fire hazards will be a part of this review.

- The staff has proposed a requirement for equipping PWRs with new water level instrumentation for midloop operation that would rely on measurement techniques not affected by pressure errors. The staff acknowledges that control of level, based on existing measurement techniques, has improved as a result of the requirements contained in GL 88-17, "Loss of Decay Heat Removal." The incremental safety improvement that would result from the addition of new water level instrumentation needs to be evaluated and contrasted with that resulting from more vigorous enforcement of the GL 88-17 requirements.

- The staff has proposed a number of technical specifications for the control of safety-related equipment during shutdown and low-power operations. NEI points out that these requirements overlap those cited in Section 50.65(a)(3) of the Maintenance Rule, which specifies that "In performing monitoring and preventive maintenance activities, an assessment of the total plant equipment taken out of service should be taken into account to determine the overall effect on the performance of plant safety functions." This section of the Maintenance Rule appears to provide the staff with the enforcement authority necessary to ensure proper control of safety-related equipment during shutdown and low-power operations. The use of such an approach also recognizes that the risk arising from shutdown and low-power operations is plant-specific in nature. Additionally, this approach would also provide licensees with more flexibility in their management of outage work.

We wish to be kept informed as development of this important issue progresses.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

References:
1. Memo dated May 2, 1994, from M. Virgilio, Office of Nuclear Reactor Regulation, to J. Larkins, ACRS, transmitting revised copy of proposed Rule and associated draft Regulatory Guide on shutdown and low-power operations
2. Memorandum dated March 14, 1994, from F. Miraglia, Office of Nuclear Reactor Regulation, for E. Jordan, Chairman, Committee to Review Generic Requirements, transmitting proposed rulemaking package on shutdown and low-power operations containing: Federal Register Notice with proposed Rule, a draft Regulatory Analysis, draft Regulatory Guide 1.XXX, "Shutdown and Low-Power Operations at Nuclear Power Plants", and NUREG-1449, "Shutdown and Low-Power Operations at Commercial Nuclear Power Plants in the United States"
3. Letter dated April 8, 1994, from R. Burski, Chairman, CE Owners Group, to J. E. Wilkins, ACRS, transmitting comments on proposed regulatory requirements for shutdown and low-power operations
4. Letter dated March 28, 1994, from W. Rasin, Nuclear Energy Institute, to E. Jordan, AEOD, transmitting comments on proposed regulatory requirements for shutdown and low-power operations
5. Memorandum dated September 20, 1993, from A. Thadani, Office of Nuclear Reactor Regulation, for J. Larkins, ACRS, transmitting "Questions from the Operations Subcommittee Regarding Shutdown and Low-Power Operations"
6. Letter dated September 15, 1992, from D. A. Ward, Chairman, ACRS, to J. M. Taylor, EDO, Subject: NRC Staff's Proposed Resolution of Issues Identified in its Evaluation of Shutdown and Low-Power Operations
7. Letter dated October 15, 1992, from J. M. Taylor, EDO, to D. A. Ward, Chairman, ACRS, Subject: NRC Staff's Proposed Resolution of Issues Found During its Evaluation of Shutdown and Low-Power Operations
8. Letter dated April 9, 1992, from D. A. Ward, Chairman, ACRS, to J. M. Taylor, EDO, Subject: Evaluation of the Risks During Shutdown and Low-Power Operations for U.S. Nuclear Power Plants

9. Letter dated August 13, 1991, from D. A. Ward, ACRS Chairman, to J. M. Taylor, EDO, Subject: Evaluation of Risks During Low-Power and Shutdown Operations of Nuclear Power Plants

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

June 14, 1994

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

Dear Chairman Selin:

SUBJECT: THERMO-LAG FIRE BARRIERS

During the 410th meeting of the Advisory Committee on Reactor Safeguards, June 9-10, 1994, we discussed the proposed staff approach for resolving Thermo-Lag fire barrier issues with representatives of the NRC staff and Nuclear Energy Institute (NEI). Our Subcommittee on Auxiliary and Secondary Systems reviewed this matter during a meeting on June 8, 1994. We also had the benefit of the documents referenced. This report is in response to the March 18, 1994 Staff Requirements Memorandum.

We agree with the staff's view that an immediate order to require upgrading of inadequate Thermo-Lag fire barriers is not needed based on defense-in-depth arguments and the fact that compensatory measures are already in place at those plants that have not resolved their Thermo-Lag problems.

In SECY-94-127, the staff describes the following four options for resolving the Thermo-Lag fire barrier issues:

Option 1 - Require Compliance with Existing NRC Fire Barrier Requirements

Option 2 - Develop Guidance for Rating Fire Barriers Based Upon a Range of Combustible Loadings for Fire Endurance Tests

Option 3 - Develop a Performance-Based Approach Using a Lead Plant

Option 4 - Develop a Performance-Based Fire Protection Rule

We support the staff recommendation described as Option 1, which includes provisions for plant-specific exemptions as permitted in the current regulations. However, we believe that exemptions under Option 1 should not be limited to those permitted by precedent.

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Fire-analysis techniques have advanced substantially since the current fire protection regulations were promulgated. These advances justify a reexamination of the bases for granting exemptions. We recommend that, in the near term, the staff and industry work toward the development of generic guidelines for using performance-based approaches to justify exemptions.

We are advocates of risk-based regulation and therefore support the staff's plan, described in SECY-94-090, to develop risk-based and performance-oriented fire protection regulations and recommend that any such regulatory framework include consideration of fire risk during shutdown conditions.

Additional comments by ACRS Member Ivan Catton are presented below.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

Additional Comments of ACRS Member Ivan Catton

While I agree with some of what is said in the above report, I do not understand why the implementation of Option 2 is considered to be so complex. The computational tools are available to support the selection of Option 2 as a means to resolve the Thermo-Lag issues without resorting to a large number of exemptions. There are examples of how this can be done. Further, most of what must be done will support the effort to achieve a performance-based fire protection regulation. I believe it is time to follow the lead of other countries (e.g., Sweden, Australia, and others) in moving toward realistic performance-based fire protection regulation.

References:

1. SECY-94-127 dated May 12, 1994, from James M. Taylor, Executive Director for Operations, NRC, for the Commissioners, Subject: Options for Resolving the Thermo-Lag Fire Barrier Issues
2. SECY-94-128 dated May 12, 1994, from James M. Taylor, Executive Director for Operations, NRC, for the Commissioners, Subject: Status of Thermo-Lag Fire Barriers
3. Memorandum dated March 18, 1994, from Samuel J. Chilk, Secretary, to J. Ernest Wilkins, Jr., ACRS Chairman, and James M. Taylor, EDO, Subject: Staff Requirements - Periodic Meeting with the ACRS, March 10, 1994

4. SECY-94-090 dated March 31, 1994, from James M. Taylor, Executive Director for Operations, NRC, for the Commissioners, Subject: Institutionalization of Continuing Program for Regulatory Improvement
5. SECY-94-024 dated February 4, 1994, from James M. Taylor, Executive Director for Operations, NRC, for the Commissioners, Subject: Resolution of Issues Concerning Thermo-Lag Fire Barriers
6. SECY-93-143 dated May 21, 1993, from James M. Taylor, Executive Director for Operations, NRC, for the Commissioners, Subject: NRC Staff Actions to Address the Recommendations in the Report on the Reassessment of the NRC Fire Protection Program
7. Memorandum dated March 25, 1994, to Holders of Operating Licenses from Luis A. Reyes, Office of Nuclear Reactor Regulation, NRC, Subject: Fire Endurance Test Acceptance Criteria for Fire Barrier Systems Used to Separate Redundant Safe Shutdown Trains Within the Same Fire Area (Supplement 1 to Generic Letter 86-10, "Implementation of Fire Protection Requirements")
8. Letter dated March 4, 1994, from Alex Marion, Nuclear Management and Resources Council, to C. McCracken, Office of Nuclear Reactor Regulation, NRC, transmitting NUMARC Industry Application Guide to Evaluate Thermo-Lag Fire Barriers (Draft D)

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

July 13, 1994

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

Dear Chairman Selin:

SUBJECT: EMERGENCY PLANNING ZONES, PROTECTIVE ACTION GUIDELINES, AND THE NEW SOURCE TERMS

During the March 10, 1994 meeting with the Commissioners, the ACRS agreed to consider the implications of the results reported in the ASEA Brown-Boveri Combustion Engineering (ABB-CE) Standard Safety Analysis Report for System 80+ design that the calculated doses for the design basis accidents (DBAs), using the new source terms and a hypothetical site, were less than protective action guidelines (PAGs) levels at the site boundary. During our 410th meeting on June 9-10, 1994, we had the benefit of a staff presentation on the use of PAGs in emergency planning. We also had the benefit of the referenced documents.

Calculated doses associated with the DBA prescription are sensitive to parameters associated with the DBA specifications, the containment design, and the site characteristics. These parameters include, for example, the source term itself (amount, timing, and chemical form), the effectiveness of engineered and natural aerosol mitigation processes (e.g., sprays and containment dimensions), containment volume and leak rate, the associated DBA pressure source, and specified meteorological conditions.

The items that appear to be major contributors to the low dose values calculated for System 80+ are:

- the large volume of the containment,
- an effective spray system design,
- an annular containment design that routes leakage through a filtered vent,
- the new specification for the source term contained in draft NUREG-1465 (particularly the timing), and

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- the use of "medium" meteorological conditions as taken from the EPRI Utility Requirements Document for a hypothetical site instead of "worst-case" conditions.

The implication of the low value of the calculated DBA dose at the site boundary is that it points to a need to revisit the technical basis and rationale that underlie the present regulatory guidance on emergency planning - particularly with respect to the extent of Emergency Planning Zones (EPZs). This is an opportunity to develop a trial application of the concept of risk-based regulations.

The existing regulations require that emergency response plans be established and the guidance calls for including provisions for sheltering and/or evacuating within a 10-mile radius (i.e., plume exposure EPZ) around the reactor site in the event that doses anywhere in that region during an accident in progress are projected to exceed the PAGs. In addition, a 50-mile ingestion pathway zone is called for such that protective measures are available in the event that projected doses exceed additional PAG values in that zone.

The rationale for these requirements seems to be defined in NUREG-0654, from which we cite the following:

> "... it would be unlikely that any protective action for the plume exposure pathway would be required beyond the plume exposure EPZ."
>
> "... the likelihood of exceeding ingestion pathway protective action guide levels at 50 miles is comparable to the likelihood of exceeding plume exposure pathway protective action guide levels at 10 miles."
>
> "Projected doses from most core melt sequences would not exceed PAGs outside the [10-mile] EPZ."
>
> "For the worst core melt sequences, immediate life threatening doses would generally not occur outside the [10-mile] EPZ."

This is a good example of the type of regulatory basis that has concerned the ACRS for years. It has the "right-sounding" words but is lacking in real substance and is inflexible for new designs. In particular, it has only a loose risk basis rooted primarily in the results from WASH-1400, is specific only for contemporary LWRs, and uses qualifiers such as "unlikely," "likelihood," "most," and "generally." We believe the regulations related to emergency planning deserve better.

We believe the current regulatory extent of the EPZs as applied to existing nuclear plants implies an underlying level of "accepted risk." If a comparable risk basis were to be applied to advanced plants, then the associated resulting EPZs would be expected to be smaller, possibly shrinking to the size of the site boundary.

The Commission, in the July 30, 1993 SRM, directed "... the staff should submit to the Commission recommendations for proposed technical criteria and methods to use to justify simplifications of existing emergency planning requirements." We support this directive from the Commission and note that, as part of the draft PRA implementation plan, the staff intends to proceed with efforts in that direction. We recommend that, as part of this effort, the staff be directed to develop firm risk-based criteria for EPZs for use with advanced plant designs. We believe developing such criteria would first require developing answers to the following questions:

- What level of risk is being "accepted" for currently operating LWRs with their existing EPZs?

- Is this level of "accepted" risk appropriate? If not, what should it be?

- For the advanced plant designs, what would be the size of the EPZs based on a level of risk comparable to the "accepted" value? What are the implications of this result?

We recognize that developing criteria based on "acceptable risk" would be conceptually as difficult as was development of the Safety Goal criteria. We also recognize that defense-in-depth might be a sufficient regulatory basis for the present extent of EPZs. Nevertheless, we believe that now is the appropriate time, and that the guidance on EPZs is the appropriate subject, for a trial effort on risk-based regulation to begin.

Sincerely,

T S Kress

T. S. Kress
Chairman

<u>References</u>:

1. Memorandum dated March 18, 1994, from Samuel J. Chilk, Secretary, to J. Ernest Wilkins, Jr., ACRS Chairman, and James M. Taylor, EDO, Subject: Staff Requirements - Periodic Meeting with the ACRS, March 10, 1994
2. U.S. Nuclear Regulatory Commission, NUREG-0396, "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants," December 1978

3. U.S. Nuclear Regulatory Commission, NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," February 1980

4. Staff Requirements Memorandum dated July 30, 1993, from Samuel J. Chilk, Secretary, for James M. Taylor, Executive Director for Operations, Subject: SECY-93-092 - Issues Pertaining to the Advanced Reactor (PRISM, MHTGR, and PIUS) and CANDU 3 Designs and Their Relationship to Current Regulatory Requirements

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

July 13, 1994

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

Dear Mr. Taylor:

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

During the 411th meeting of the Advisory Committee on Reactor Safeguards, July 7-8, 1994, we completed our discussion related to the results of our recent reviews of the General Electric Nuclear Energy (GENE) Advanced Boiling Water Reactor (ABWR) (Reference 1) and the ASEA Brown-Boveri Combustion Engineering (ABB-CE) System 80+ (Reference 2) applications for design certification from the perspective of potential areas for staff action for operating nuclear power plants and the review of future plant designs. These reviews provided us with an opportunity to consider present regulatory practices and procedures vis-a-vis the "state-of-the-art" design requirements for these evolutionary light water reactors (ELWRs).

The following are some issues that we believe the staff should address as Generic Issues, as Technical Specification Improvement Program issues, as revisions to the Standard Review Plan, or as additional research needs.

1. Turbine Inspection Requirements - In the course of reviewing the potential for turbine rotor failure related to the ABWR and System 80+ designs, we learned that the staff has not prepared an appropriate set of preoperational and inservice inspection, evaluation and acceptance requirements for turbine rotor, other than those employing shrunk-on disks.

   Some current licensees have replaced, or are planning to replace, shrunk-on disk rotors with rotors of a different design. We believe that the staff should develop appropriate positions for the various designs on a priority basis.

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2. Technical Specification Requirements for Onsite Power Sources - In our letter to you dated February 17, 1994, concerning three issues relating to the 10 CFR Part 52 design certification process for ALWRs, we recommended that the staff resolve the matter of credit for ELWR alternate AC sources when 1E emergency diesel generators are out of service during power operation. We suggested that Technical Specification requirements for such onsite power sources be based on appropriate probabilistic considerations. Subsequently, ABB-CE requested such credit for System 80+ and the staff has granted an allowable outage time for a 1E emergency diesel generator of up to 14 days when the combustion turbine-generator is available. We now recommend that the staff expand this concept to include operating nuclear power plants.

   It is our understanding that Technical Specification requirements for onsite power sources will be incorporated into the Shutdown and Low Power Operations Rule.

3. Reactor Water Cleanup System Safety - The Reactor Water Cleanup (RWCU) System is of safety concern for boiling water reactor plants because it is a high-energy, non-safety system, portions of which may be located inside of the secondary containment. The secondary containment also houses numerous engineered safety features and the Fuel Pool Cooling System. For operating plants, the RWCU System supply line from the reactor vessel is usually a 6-inch pipe. A rupture of this pipe inside of the secondary containment results in a loss of reactor coolant which may create a serious environmental disruption throughout the secondary containment before it can be isolated.

   An ACRS staff report (Reference 3) identified a number of safety-related deficiencies in a similar system for the ABWR. Subsequently, GENE developed a requirement for environmental qualification of all safety-related components and the Fuel Pool Cooling System inside of the secondary containment. The qualification was based mostly on the adverse atmosphere created before complete closure of the isolation valves following a supply line pipe break. Generally, operating plants do not provide a comparable level of environmental qualification.

   Another GENE change was the addition of a second isolation valve in the supply line inside of the primary containment. This valve isolates the reactor vessel from the supply line pipe break in the event that isolation is not achieved by closing the two primary containment isolation valves under blowdown flow conditions. The added valve is not capable of blowdown isolation. It is closed by manual actuation after the blowdown is completed, thereby achieving reactor vessel isolation and interruption of any prolonged release of Emergency Core Cooling System (ECCS) water to the break which is outside of primary containment. Operating plants may not have a similar capability. We recommend that this issue be investigated for operating BWRs.

4. Review of Chilled-Water Systems - A number of operating plants use large Chilled-Water Systems to provide essential environmental cooling. Because there is no Standard Review Plan (SRP) for these systems, the staff has used other guidance such as SRP 9.2.2 (Reactor Auxiliary Cooling Water Systems) when evaluating the safety of such systems. However, this guidance is not appropriate for the evaluation of refrigeration systems.

   In determining plant safety, the NRC staff needs to evaluate the performance of Chilled-Water Systems under various accident heat loads and during loss-of-offsite-power events, and to consider the ability of such systems to restart and function after tripping or after a prolonged station blackout. We urge that the staff develop better guidance and positions with which to enhance the scope and quality of its plant reviews of Chilled-Water Systems.

5. Filters or Water Separators for the Hardened Vents Installed on Operating BWR Containments - A great deal of analysis was done to demonstrate that the ABWR Containment Overpressure Protection System is adequate without filters or water separators. We are not aware that such an analysis has been done for those operating BWRs with hardened vents. We believe their need for filters or water separators should be reevaluated.

6. Fuel-Coolant Interactions - We are concerned that the safety case with respect to fuel-coolant interactions is based mostly on arguments of low probability of occurrence. It concerns us that neither the industry nor the NRC staff is able to predict limits to the energetics (below purely thermodynamic limits) based on either first principles or sufficient empirical evidence. We believe additional research is needed on this issue.

7. Adequacy and Use of PRA - We are concerned that there are no clear regulatory criteria for what constitutes an acceptable PRA. By accepting the PRAs which have already been submitted, the staff is essentially establishing the regulatory criteria by precedent rather than by promulgating specific requirements. We believe consideration should be given to establishing minimum requirements for PRAs.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

References:

1. ACRS Report dated April 14, 1994, from J. Ernest Wilkins, Jr., ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: Report on Safety Aspects of the General Electric Nuclear Energy Application for Certification of the Advanced Boiling Water Reactor Design
2. ACRS Report dated May 11, 1994, from T. S. Kress, ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: Report on the Safety Aspects of the ASEA Brown Boveri-Combustion Engineering Application for Certification of the System 80+ Standard Plant Design
3. Advisory Committee on Reactor Safeguards Report by S. E. Mays and M. E. Stella, "ABWR Reactor Water Cleanup System Review," July 30, 1992

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

July 14, 1994

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

Dear Chairman Selin:

SUBJECT: PROPOSED NATIONAL ACADEMY OF SCIENCES/NATIONAL RESEARCH COUNCIL STUDY AND WORKSHOP ON DIGITAL INSTRUMENTATION AND CONTROL SYSTEMS

During the 411th meeting of the Advisory Committee on Reactor Safeguards, July 7-8, 1994, we discussed the proposal by the National Academy of Sciences/National Research Council (NAS/NRC) for a study and workshop on the "Application of Digital Instrumentation and Control Technology to Nuclear Power Plant Operations and Safety." During our review, we had the benefit of discussions with representatives of the NRC staff and the NAS/NRC. We also had the benefit of the documents referenced. This report is in response to a Commission request in the March 18, 1994 Staff Requirements Memorandum.

The proposal focuses primarily on hardware and software issues that arise from the introduction of digital instrumentation and control (I&C) technology in nuclear power plants. Human factors considerations appear to be limited to human-machine interface issues related directly to digital technology. We believe this balance in emphasis is proper. The issues associated with hardware and software are very broad and any significant diversion of effort from these issues is undesirable. In addition, we believe that the staff's Human Factors Engineering Program Review Model and the acceptance criteria used for evolutionary reactors provide reasonable regulatory guidance for human factors issues. The current need is for a corresponding regulatory framework for hardware and software issues associated with digital I&C technology.

We believe the NAS/NRC study panel findings will assist the Commission in providing necessary guidance to the staff for the development of a regulatory framework for digital I&C. While the staff and the ACRS have identified a number of concerns that are believed to be significant, the ACRS strongly urges that the study panel be permitted to select the issues to be considered.

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We expect that the NAS/NRC study will make use of knowledge that has been developed in other industries with digital system experience. We are particularly interested in the state-of-the-art of the development of software specifications, verification and validation of software, the potential vulnerabilities of hardware over the spectrum of adverse environments which can occur in nuclear power plants, and the prediction of reliability (including common-mode failure).

We recommend that the staff identify in the background papers provided to the NAS/NRC study panel those applicable NRC regulations, IEEE standards, Electric Power Research Institute Utility Requirements, and vendor information that pertain to safety-related digital I&C system development.

We understand that a visit to the NRC Technical Training Center simulators is planned. It may be more useful for study panel members to visit a nuclear plant digital system vendor to observe developmental mock-ups and to discuss nuclear power plant digital I&C designs. Consideration should also be given to visiting an operating plant that employs digital control and protection systems.

We look forward to meeting with members of the study panel during the course of the study.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

<u>References</u>:

1. Memorandum dated March 18, 1994, from Samual J. Chilk, Secretary, to J. Ernest Wilkins, Jr., ACRS Chairman, and James M. Taylor, EDO, Subject: Staff Requirements - Periodic Meeting with the ACRS, March 10, 1994
2. Memorandum dated March 1, 1994, from James M. Taylor, Executive Director for Operations, NRC, for The Commission, Subject: Nuclear Safety Research Review Committee Report Dated January 14, 1994
3. Memorandum dated May 3, 1994, from James M. Taylor, Executive Director for Operations, NRC, for the Commission, Subject: Staff Response to Nuclear Safety Research Review Committee Reports Dated January 14 and February 16, 1994
4. ACRS Letter Report dated March 18, 1993, from Paul Shewmon, ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: Computers in Nuclear Power Plant Operations
5. ACRS Letter Report dated November 16, 1993, from J. Ernest Wilkins, Jr., ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: Computers in Nuclear Power Plant Operations

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

September 12, 1994

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

Dear Chairman Selin:

SUBJECT: PROPOSED GENERIC LETTER 94-XX, "VOLTAGE-BASED REPAIR CRITERIA FOR WESTINGHOUSE STEAM GENERATOR TUBES"

During the 412th meeting of the Advisory Committee on Reactor Safeguards, August 4-5, 1994, we reviewed the subject generic letter (GL), an associated differing professional opinion (DPO), and a draft of an Advance Notice of Proposed Rulemaking on Steam Generator Tube Integrity. During the 413th meeting, September 8-10, 1994, we discussed the NRC staff's revised calculations for radiological consequences of a main steamline break associated with a degraded steam generator. During our review, we had the benefit of discussions with representatives of the NRC staff and the Nuclear Energy Institute (NEI), as well as the author of the DPO. We also had the benefit of the documents referenced. In part, this report is in response to a request made by the Executive Director for Operations in a July 15, 1994, memorandum to the Executive Director of the Advisory Committee on Reactor Safeguards.

Although existing mechanics-based design criteria and evaluation methods have served to ensure adequate steam generator tube integrity, they appear to be overly conservative for some types of degradation, and result in unnecessary tube plugging or repair. The proposed GL provides an alternate approach applicable solely to axially oriented outside diameter stress corrosion cracking (ODSCC) of tubes at the tube-support-plate intersections in Westinghouse steam generators with drilled-hole support plates.

We support the issuance of the proposed GL for public comment. We have reviewed the DPO and do not believe that it identifies any fundamental shortcomings in the approach proposed in the GL.

The DPO cites a high core damage frequency (CDF) of 3.4×10^{-4}/RY. This value was based on a preliminary scoping analysis performed by the Office of Nuclear Regulatory Research (RES). Subsequent analyses performed by RES in support of the application of the interim plugging criteria for the Trojan Nuclear Plant and for NUREG-1477 give CDFs of less than 2×10^{-6}/RY. These values are based on conservative estimates of leakage from degraded tubes. Except perhaps for steamline breaks, the structural restraint provided by the tube-support plate provides a high degree of assurance against tube bursts.

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The criticism in the DPO of the approach used in the proposed GL and in the Standard Review Plan to compute radiological releases during a main steamline break appears to warrant further consideration. The basis for the definition of the iodine spike during a rapid depressurization transient as 500 times the equilibrium release rate is not clear. However, an alternate calculation of the release based on the gap inventory of iodine in leaking fuel elements appears to give comparable releases. In both approaches there appears to be margin in meeting the 10 CFR Part 100 limits. The staff should review the spiking data or consider other approaches to estimate the iodine release to provide a more satisfactory basis for the radiological dose estimates. In particular, we encourage the staff to quantify the level of conservatism in its analyses.

While the proposed GL appears to provide a useful interim approach for assessing steam generator tube integrity, the database for the present empirical correlations for burst pressure and leakage with the bobbin coil voltage, appears to be only marginally adequate, and more data need to be developed.

The use of such empirical correlations as the basis for assuring the integrity of steam generator tubing would also seem to require an ongoing tube-pull program with associated burst and leak testing and metallurgical examinations as outlined in the proposed GL to ensure that the correlations remain valid as degradation continues. In the longer term, it would be worthwhile to reconsider a fracture-mechanics-based approach utilizing improved non-destructive examination techniques that provide more accurate detection and characterization of degradation. Ongoing efforts in RES and in industry to develop and implement such an approach should be continued and encouraged.

We agree with the staff position that rulemaking is the preferred regulatory approach to the problem of steam generator tube degradation, although we are skeptical that a new rule can be developed as expeditiously as the proposed schedule suggests. The overall objective and attributes of the new rule, as described by the staff, pay proper obeisance to performance-based regulation. We would like to be kept informed of the progress by the staff in the implementation of a performance-based approach.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

References:
1. Memorandum dated July 8, 1994, from F. J. Miraglia, Deputy Director, Office of Nuclear Reactor Regulation, for E. L. Jordan, Chairman, Committee to Review Generic Requirements, Subject: CRGR Review of Generic Letter 94-XX, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes"
2. Memorandum dated July 15, 1994, from J. M. Taylor, NRC Executive Director for Operations, for J. T. Larkins, ACRS Executive Director, Subject: ACRS Review of Proposed Generic Letter 94-XX, Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes
3. U.S. Nuclear Regulatory Commission, 10 CFR Part 50, RIN 3150-, Steam Generator Tube Integrity (7590-01), Draft Advance Notice of Proposed Rulemaking, received July 20, 1994
4. Memorandum dated August 17, 1994, from J. A. Calvo, NRC Office of Nuclear Reactor Regulation, for J. T. Larkins, ACRS Executive Director, Subject: Revisions to Slides Used by Staff During August 3, 1994, Subcommittee Briefing on Steam Generator Alternate Repair Criteria
5. U.S. Nuclear Regulatory Commission, NUREG-1477, "Voltage-Based Interim Plugging Criteria for Steam Generator Tubes," Draft Report for Comment, June 1993
6. Memorandum dated January 15, 1993, from E. S. Beckjord, Director, Office of Nuclear Regulatory Research, to T. E. Murley, Director, Office of Nuclear Reactor Regulation, Subject: Interim Plugging Criteria for Trojan Nuclear Plant

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

September 14, 1994

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

Dear Mr. Taylor:

SUBJECT: PROPOSED GENERIC LETTER ON THE USE OF NUMARC/EPRI REPORT TR-102348, "GUIDELINE ON LICENSING DIGITAL UPGRADES"

During the 413th meeting of the Advisory Committee on Reactor Safeguards, September 8-10, 1994, we reviewed the subject proposed generic letter. During our review, we had the benefit of discussions with representatives of the NRC staff and the Nuclear Energy Institute. We also had the benefit of the documents referenced.

The proposed generic letter endorses, with two clarifications, Nuclear Management and Resources Council/Electric Power Research Institute (NUMARC/EPRI) Report TR-102348 as useful guidance for effectively implementing digital upgrades and for determining when these can be performed without prior NRC staff approval under the requirements of 10 CFR 50.59.

We basically concur with the proposed generic letter and have no objection to issuing it for public comment. However, we believe that additional clarification should be provided regarding equipment environmental compatibility. Specifically, it should be made clear in the generic letter that the environmental requirements as defined in Subsection 5.3, "Compatibility With the Environment," of the NUMARC/EPRI report include all environmental conditions resulting from internal and external events to which the equipment may be subjected. This subsection currently focuses on the need to address electromagnetic interference. We believe that any guideline which purports to cover environmental compatibility issues for replacement equipment must require that other environmental stressors such as temperature, humidity, radiation, vibration/seismic, and smoke be addressed. We note that the need to prioritize these and to verify the appropriateness of current research programs was identified in our letter of November 12,

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1992, and that you agreed. We anticipate a briefing on the results of this effort.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

<u>References</u>:

1. Memorandum dated August 30, 1994, from E. Doolittle, NRC Office of Nuclear Reactor Regulation, to J. Larkins, ACRS Executive Director, forwarding Proposed NRC Generic Letter on the Use of NUMARC/EPRI Report TR-102348, "Guideline on Licensing Digital Upgrades"
2. Letter dated December 22, 1993, from W. Rasin, NUMARC, to W. Russell, NRC Office of Nuclear Reactor Regulation, forwarding EPRI Report TR-102348
3. ACRS letter dated November 12, 1992, from Paul Shewmon, ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: Environmental Qualification for Digital Instrumentation and Control Systems
4. Letter dated December 10, 1992, from James M. Taylor, NRC Executive Director for Operations, to Paul Shewmon, ACRS Chairman, Subject: Environmental Qualification for Digital Instrumentation and Control Systems

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

September 14, 1994

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

Dear Chairman Selin:

SUBJECT: REVISED REGULATORY ANALYSIS GUIDELINES

During the 413th meeting of the Advisory Committee on Reactor Safeguards, September 8-10, 1994, we discussed the proposed final "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission." During this meeting, we had the benefit of discussions with representatives of the NRC staff. We note that the industry did not have the opportunity to review the staff response to public comments. We had provided comments on a preliminary version of these Guidelines to the Executive Director for Operations in a letter dated November 12, 1992. We also had the benefit of the documents referenced.

In our November 12, 1992 letter, we made a number of substantive comments on areas in which we disagreed with the staff proposals. In the revised version, the staff has satisfactorily addressed most of our earlier concerns. In addition, we believe the staff response to the public comments has been balanced and appropriate.

We believe these Guidelines will be valuable to the NRC staff in its various decision-making functions. At this time, we still have concerns in two areas:

1. Until new guidance has been developed on the appropriate monetary values to apply to adverse health and land contamination effects, the staff proposes the continued use of an undiscounted $1000/man-rem. as a surrogate for the actual discounted values.

   We do not support this proposal. The correct treatment requires separate, realistic values for each effect and these should be discounted for present-worth evaluation. The Guidelines should not be issued until a technically correct approach with the appropriate values is developed.

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2. The revised Guidelines now propose a definition for containment failure that is "... consistent with the performance goal used in the review of evolutionary ALWRs and documented in SECY-93-087." This is a change from the definition used in a prior version of the Guidelines which was taken from NUREG-1150.

   The definition in NUREG-1150, which addresses the risk dominant sequences, is the appropriate one for use in these Guidelines.

The issuance of the new Regulatory Analysis Guidelines should be delayed until these issues are reconsidered.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

References:
1. Letter dated June 29, 1994, from C. J. Heltemes, Jr., NRC Office of Nuclear Regulatory Research, to T. S. Kress, ACRS Chairman, transmitting draft SECY Paper: Regulatory Analysis Guidelines of the U.S. NRC (Draft Predecisional)
2. Letter dated November 12, 1992, from Paul Shewmon, ACRS Chairman, to James M. Taylor, NRC Executive Director for Operations, Subject: Revised Regulatory Analysis Guidelines

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

September 19, 1994

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

Dear Chairman Selin:

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

During the 413th meeting of the Advisory Committee on Reactor Safeguards, September 8-10, 1994, we reviewed the proposed revisions to Appendix J to 10 CFR Part 50, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." Our Subcommittee on Containment Systems also reviewed this matter at a meeting on September 7, 1994. During this review, we had the benefit of discussions with representatives of the NRC staff, Nuclear Energy Institute (NEI), Grand Gulf Nuclear Station (Entergy Operations, Inc.), and ANS-56.8 Working Group (Containment System Leakage Testing Requirements). We also had the benefit of the documents referenced.

We are in general agreement with the proposed revisions to Appendix J and have no objection to the publication of the proposed rule for public comment. The changes proposed do not appear to have significant potential to increase public risk and, in fact, may reduce risk by decreasing the probability of accidents during shutdown. In addition, the changes will permit staff and industry resources to be redirected to more risk-significant issues.

The staff identified two issues that remain unresolved with industry. These are: (1) the proposed rule allows a maximum interval for leakage testing of Type C components (isolation valves) of 60 months, whereas industry would prefer a staggered test program leading to a maximum of 120 months; and (2) the staff proposes that certain leak testing provisions be incorporated into the technical specifications for the individual plants, whereas the industry proposes that the leak testing provisions be a commitment in the Final Safety Analysis Report (FSAR).

With regard to the leakage testing interval for Type C components, the arguments for the 120-month interval are reduction in costs, in occupational exposure, and in shutdown risks. The staff arguments

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for an initial 60-month limit are: (1) a conservative approach should be adopted until experience is gained, and (2) aging effects on leakage may escape timely detection if a period longer than 60 months is allowed. We accept the staff position on this issue, which includes the option for a 120-month interval after evaluating experience with the proposed rule. Our acceptance is conditional on the assumption that valve operability (as opposed to leakage) will be demonstrated appropriately by other means such as those already implemented under Generic Letter 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance."

We note that any shutdown risk benefit that may be gained by increasing the test interval has not been quantified. In addition, the staff has acknowledged that it has not looked for aging effects on valve leakage in older plants. We recommend that the staff examine both of these issues in order to provide additional insights relative to the appropriate maximum test interval for Type B and C components. The shutdown risk issue could be evaluated by extension of the recently completed shutdown risk assessments for Surry and Grand Gulf nuclear plants.

With respect to the second unresolved issue, both the staff and NEI agree that the allowable leakage rate for the containment (which we view as the performance goal) should be included in the Technical Specifications (TS). The staff is still considering requirements that may be needed in the TS to ensure that program changes are reviewed by the staff. An example is the algorithm to be used for extension of Type C isolation valve leakage testing. NEI argues that it is sufficient to place these requirements in the FSAR so that changes can be made using the 10 CFR 50.59 process. Since the additional TS requirements proposed by the staff are counter to the concept of the performance-based Maintenance Rule, we recommend that the staff adopt the NEI position on this issue.

We plan to review this matter after reconciliation of the public comments.

Additional comments by ACRS Members Thomas S. Kress and Robert L. Seale and ACRS Members James C. Carroll, Ivan Catton, and William J. Lindblad are presented below.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

Additional Comments by ACRS Members Thomas S. Kress and Robert L. Seale

We fully agree with the Committee that there is unlikely to be an unacceptable increase in risk as a result of this proposed change to the leakage testing interval and that this is an appropriate area to provide some regulatory relief for the industry. Nevertheless, we have two objections to the form of the proposed revisions:

1. We believe a bad precedent is set for performance-based regulations by having the relaxation (or tightening) of the regulatory oversight be on the performance measure frequency itself. It should be a general principle that these be separate.

2. We are unconvinced that an adequate technical basis has been established that two consecutive successful leakage tests provide appropriate criteria for acceptable performance in this case. This, again, sets a bad precedent for supposedly performance-based regulations.

Additional Comment by ACRS Members James C. Carroll, Ivan Catton, and William J. Lindblad

While we believe the Appendix J revisions proposed by the staff will protect public health and safety, the further provisions that were proposed by NEI (staggered testing of classes of Type C components with a maximum testing interval of 120 months) seem to us to be proper as well. The conditions under which extended test intervals would be permitted appear to be consistent with those contemplated by the Maintenance Rule.

References:
1. Memorandum dated August 23, 1994, from Joseph A. Murphy, Office of Nuclear Regulatory Research, NRC, for John T. Larkins, Executive Director, Advisory Committee on Reactor Safeguards, Subject: Performance-Based Containment Leakage Test Rulemaking (Transmitting Draft SECY Paper for the Commissioners from James M. Taylor, EDO, undated)
2. Nuclear Energy Institute, NEI 94-01, Draft Revision C, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," August 1, 1994
3. U.S. Nuclear Regulatory Commission, NUREG-1493, Draft (Revision 2, 3/31/94), "Performance-Based Containment Leak-Test Program"
4. Electric Power Research Institute/Science Applications International Corporation, EPRI TR-104285, Final Report dated August 1994, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals"

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

September 20, 1994

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

Dear Chairman Selin:

SUBJECT: PROPOSED FINAL VERSION OF NUREG-1465, "ACCIDENT SOURCE TERMS FOR LIGHT-WATER NUCLEAR POWER PLANTS"

During the 413th meeting of the Advisory Committee on Reactor Safeguards, September 8-10, 1994, we discussed the proposed final version of NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants." During the meeting, we had a discussion with the staff regarding how comments on the draft version of this document have been accommodated in the final version. We also had a presentation by a representative of Northeast Utilities on the safety importance of adopting proposed accident source term timing assumptions. The draft version was discussed with the Committee at the 381st meeting in January 1992, and comments were provided in our report dated January 15, 1992. We also had the benefit of the documents referenced.

NUREG-1465 defines accident source terms for use in the safety analysis of future light water reactors to replace the source term specified in Regulatory Guides 1.3 and 1.4. The proposed source terms are based on the vast amount of research sponsored over the last 15 years by the NRC and others. The proposed source terms specify the releases of eight categories of radionuclides over four time intervals after the initiation of an accident. Most of these radionuclides are expected to form aerosol particles in the containment. Only the noble gases and 5 percent of the iodine are in gaseous form. This contrasts with the source term now used which specifies an instant release consisting of 100 percent of the core inventory of the noble gases and 50 percent of the iodines (half of which are assumed to deposit on interior surfaces very rapidly) to the containment.

We believe it is important to have more realistic accident source terms available for regulatory activities. NUREG-1465 presents source terms which are a vast improvement over the source term now available. We do, however, have some comments.

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A variety of calculations has been examined to develop the proposed source terms. In some cases, bounding values determined from these calculations have been adopted. In other cases, mean values have been selected, and in still others, values less than the mean have been chosen. As a result, it is difficult to ascertain the conservatism inherent in the proposed source terms. We believe it important to clarify this level of conservatism especially since the proposed source terms may be used for the analyses of both design basis and beyond design basis accidents. Appropriate levels of conservatism are quite different for these two classes of accidents.

Release fractions of some categories of radionuclides have been adjusted in the final version of NUREG-1465 from values in the draft that were derived from calculations. It appears that these adjustments have been based on expert opinions provided in comments by reviewers of the draft report. We believe these adjustments need to be better justified or not be made.

Ongoing source term research activities may yield results that would substantially alter the understanding that has been the basis of the proposed source terms. A mechanism is needed for timely updating of regulatory source terms in response to significant research findings.

The target application of the proposed source terms is to future light water reactors. Since the source terms have been derived from calculations for existing light water reactors, explicit provisions should be included in NUREG-1465 to accommodate specific features of future reactors.

We agree that licensees of existing reactors should not be required to adopt the proposed source terms. Information provided to the Committee suggests that use of realistic timing assumptions for radionuclide releases to the containment during accidents can lead to safety improvements in existing plants. We urge that the risk implications be evaluated and consideration be given to allowing current licensees the option of using the timing assumptions in the proposed source terms without performing a complete source term reanalysis.

We emphasize the importance of realistic source terms in regulatory applications and believe that the use of realistic source terms could result in changes in reactor design and operation that reduce risk. We continue to be interested in the future application of the proposed source terms to specific regulatory areas and issues and wish to be kept informed.

Dr. Thomas S. Kress did not participate in the Committee's deliberations regarding this matter.

Sincerely,

W. J. Lindblad
Vice-Chairman

References:
1. Memorandum dated August 5, 1994, from Themis P. Speis, RES, for John T. Larkins, ACRS, transmitting Draft Final NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants"
2. Letter dated April 29, 1994, from J. F. Opeka, Connecticut Yankee Atomic Power Company/Northeast Nuclear Energy Company, to Mr. W. T. Russell, Director, NRR, Subject: Accident Source Term Timing Assumptions
3. Report dated January 15, 1992, from David A. Ward, Chairman, Advisory Committee on Reactor Safeguards, to Ivan Selin, Chairman, NRC, Subject: Proposed 10 CFR Part 50 and Part 100 (Nonseismic) Rule Changes and Proposed Update of Source Term

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

October 14, 1994

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

Dear Chairman Selin:

SUBJECT: POTENTIAL FOR BWR ECCS STRAINER BLOCKAGE DUE TO LOCA GENERATED DEBRIS

During the 414th meeting of the Advisory Committee on Reactor Safeguards, October 6-7, 1994, the Committee was briefed by the NRC staff on the emergency core cooling system (ECCS) recirculation strainer blockage issue raised by the event that occurred at the Barsebäck plant in Sweden on July 28, 1992. We heard previous briefings in January 1993, July 1993, and April 1994. During the present meeting, the staff discussed (1) a proposed Revision 2 to Regulatory Guide (RG) 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," (2) the contractor draft report NUREG/CR-6224, "Parametric Study of the Potential for BWR ECCS Strainer Blockage Due to LOCA Generated Debris," which has been issued for public comment, and (3) the staff plan for issuing a generic letter on this matter in August 1995. A representative of the Boiling Water Reactor Owners Group (BWROG) presented industry views and actions. We also had the benefit of the documents referenced.

The Barsebäck event involved BWR ECCS strainer blockage caused, in this case, by debris dislodged as a result of inadvertent safety valve discharge into the drywell. Our assessment of this event indicates that strainer blockage due to accident generated debris is an important safety issue for at least some BWRs and that strainer blockage was not adequately addressed in the 1985 resolution of Unresolved Safety Issue (USI) A-43, "Containment Emergency Sump Performance." The present version of RG 1.82, which formed the basis for resolution of USI A-43, deals principally with PWR ECCS sumps and provides prescriptive detailed information for PWR designs acceptable to the staff (design sketches, dimensions, etc.). The staff apparently plans to provide similarly prescriptive design information for BWR suppression pool ECCS suction strainers through its planned revision to RG 1.82.

Both the staff and BWROG agree that this is a compliance issue. However, BWR licensees may be reluctant to make plant modifications

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(beyond those interim compensatory measures required by NRC Bulletin 93-02 and its supplement) until the staff completes its deliberations on the revision to RG 1.82. Some obvious actions that licensees could have taken after the Barsebäck event to protect against the effects of LOCA generated debris are: (1) replacement of fibrous insulation with reflective metallic insulation, (2) installation of strainers with larger screen areas or other improvements, (3) installation of differential pressure sensors on ECCS pump suction strainers to detect strainer blockage, and (4) installation of strainer cleaning systems. It is our understanding that most European operators of BWRs have made or are making some or all of these modifications.

We question whether the approach the staff is taking will result in timely corrective actions. It seems to us that the onus should have been on the BWR licensees to evaluate the vulnerability of their plants to ECCS strainer blockage due to LOCA generated debris and to propose appropriate plant-specific modifications to deal with the issue. The survey performed by the BWROG in 1992 indicated that each plant is unique with respect to the nature of and potential for debris generation and strainer design and backflush capability. Therefore, plant-specific solutions are needed.

Draft NUREG/CR-6224, which was not initiated until September 1993, provides valuable insights and confirms quantitatively much that was qualitatively known and understood shortly after the Barsebäck event. A troubling insight among these is the indication that ECCS strainer blockage contributes significantly to core damage frequency (CDF) for the reference plant and similar BWRs. However, the authors of the report point out that there are many limitations and uncertainties associated both with the analysis that led to the reference plant results and with extrapolating these results to other BWRs.

Three comments evolved from our review. First, we are concerned by the implications of the prediction that the contribution due solely to strainer blockage is over three times the CDF represented in the reference plant Individual Plant Examination (IPE). We encourage the staff to examine the treatment of LOCA generated debris in other plant IPEs.

Second, we believe that the scope of draft NUREG/CR-6224 should be expanded to look at debris generation resulting from the flow of steam/water mixtures at some distance from the LOCA break location. This flow and pressure may dislodge pipe insulation, particularly if pressure equilibration is slow across the insulation, and may damage other debris producing targets such as the very large containment air handling units in the drywell.

Third, there is the potential for damaging ECCS pump seals or causing a loss of bearing cooling due to LOCA-generated fibrous and/or particulate matter. It is our understanding that most or all operating BWRs use pump discharge water for seal injection and bearing cooling. This issue, which we first raised in our letter of September 16, 1985, to the NRC Executive Director for Operations (EDO), has been discussed with the staff during our recent series of meetings. We believe that this issue needs to be evaluated and resolved as a part of the resolution of the ECCS strainer blockage issue.

In summary, we are concerned by the slow pace at which this important safety issue is being addressed. We recommend that the EDO and his senior staff critically review the current action plan and take the necessary steps to facilitate prompt resolution.

We plan to continue to monitor the NRC staff and industry's resolution of this issue.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

References:

1. Memorandum dated August 26, 1994, from Joseph A. Murphy, Office of Nuclear Regulatory Research, NRC, to Gary M. Holahan, Office of Nuclear Reactor Regulation, NRC, Subject. Review of DG-1038, Proposed Revision 2 to RG 1.82, "Water Sources for Long-Term Recirculation Cooling Following A Loss-of-Coolant Accident"
2. U. S. Nuclear Regulatory Commission, NUREG/CR-6224, Draft Report for Comment, "Parametric Study of the Potential for BWR ECCS Strainer Blockage Due to LOCA Generated Debris," August 4, 1994
3. U. S. Nuclear Regulatory Commission, OMB No. 3150-0011, NRCB 93-02: Debris Plugging of Emergency Core Cooling Suction Strainers, May 11, 1993
4. Letter dated September 16, 1985, from David A. Ward, Chairman, ACRS, to William J. Dircks, Executive Director for Operations, NRC, Subject: ACRS Review of Proposed Resolution for USI A-43, "Containment Emergency Sump Performance" and Regulatory Guide 1.82, Revision 1, "Water Sources for Long Term Recirculation Cooling Following a Loss of Coolant Accident"

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

November 10, 1994

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 PROGRAMS IN SUPPORT OF AP600 AND SBWR ADVANCED LIGHT WATER REACTOR PASSIVE PLANT DESIGN CERTIFICATION REVIEWS

During the 414th and 415th meetings of the Advisory Committee on Reactor Safeguards, October 6-7 and November 3-4, 1994, we discussed the confirmatory test and analysis programs being conducted by the Office of Nuclear Regulatory Research (RES) in support of the design certification reviews for the Westinghouse AP600 and GE Nuclear Energy (GENE) Simplified Boiling Water Reactor (SBWR) advanced light water reactors. During these meetings, we had the benefit of discussions with representatives of RES. Our Subcommittee on Thermal Hydraulic Phenomena held a meeting on August 25-26, 1994, to discuss this matter. We also had the benefit of the documents referenced.

In the absence of a full-scale test facility, an understanding of the thermal hydraulic behavior of a passive plant design will depend on the use of computer codes. The NRC staff has decided to modify RELAP5/MOD3 for its confirmatory thermal hydraulic analysis of the AP600 and SBWR designs. The important phenomena the code must simulate should be delineated in the Phenomena Identification and Ranking Table (PIRT), thus allowing one to formulate integral and separate effects experiments that will yield appropriate data for code validation. Code validation should be an integrated process involving code development, experimentation, and an understanding of the physics of two-phase flow and heat transfer.

The major objective of the thermal hydraulic code development effort should be to produce a code capable of predicting the behavior of a full-scale nuclear power plant with acceptable uncertainties. For existing nuclear plant designs, we have had the benefit of many integral and separate effects experiments at a wide variety of scales to help arrive at an estimate of the uncertainties in the code predictions. We are now dealing with two passive plant designs which evidence more complex thermal hydraulic system

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dynamics, and for which there is a paucity of relevant experimental data. There are several causes for this more complex dynamic behavior: (1) steam condensation at low pressure, (2) use of gravity-driven coolant injection, and (3) the existence of many components and complex hydraulic paths that give the system many degrees of freedom. Understanding this dynamic behavior requires evaluation of scale distortion effects and dynamic characteristics in the various test facilities. In this regard, two questions should be addressed and resolved: (1) is the evolution of a particular transient influenced by configurational and/or scale distortions, and (2) do configurational and/or scale distortions in the various test facilities preclude simulation of some important dynamic effects while introducing other dynamic effects that may not be important in a full-scale plant design? To address these questions, a top-down scaling analysis must be performed.

The NRC staff has test and analysis programs under way to address issues arising during its evaluation of the AP600 and the SBWR designs. The AP600 evaluation will be supported by testing at the Japan Atomic Energy Research Institute ROSA-V facility and the use of RELAP5/MOD3. The SBWR evaluation will be supported by testing at the Purdue University PUMA facility and the use of RELAP5/MOD3. We believe that the use of RELAP5/MOD3 for both AP600 and SBWR simulations will lead to the development of a more robust computational tool. Both programs are discussed below and some comments about the technical direction of these programs are provided.

AP600 Program

The PIRT in support of the AP600 analysis has not yet been completed. There is no indication that a PIRT was utilized for allocating resources, for assigning test objectives, or for developing the test matrices. It is necessary to complete the PIRT and confirm it on the basis of relevant scaling groups. To ensure that RELAP5/MOD3 can simulate the high ranking phenomena, specific tests in the test matrix should be associated with the high ranking phenomena in the PIRT. By doing this, all important phenomena will be addressed.

The PIRT and a proper scaling analysis for the AP600 would cover all test facilities for AP600. Unfortunately, the scaling efforts conducted for the OSU, SPES, and ROSA-V test facilities were not coordinated. The global scaling of the AP600 design, including consideration of the dynamic interactions between the major system components (pressure vessel, core makeup tank, pressurizer, steam generators, passive residual heat removal system, and accumulators), was omitted. Depressurization is not scaled, even though the methods for doing so are known. The scaling analysis for OSU, while still incomplete, could serve as a model for ROSA and SPES.

Direct counterpart tests in ROSA, OSU, and SPES are not possible. This makes it difficult to extrapolate the observed thermal hydraulic behavior to full scale. A well-planned effort to integrate experiments with code improvement and assessment is needed to quantify uncertainties. At present, RELAP5/MOD3 predicts strong oscillations both when they are observed in tests and when they are not. Consequently, the calculated behavior can neither be attributed conclusively to numerical nor physical effects. The mechanisms by which the various observed modes of oscillation are initiated and maintained need to be understood so that their potential influence on the thermal hydraulic behavior of the AP600 can be evaluated. The judicious selection of test conditions for the facilities, together with the conduct of a careful data analysis and scaling, should provide a satisfactory solution.

The demonstrated propensity for condensation oscillation events in the AP600 points to a need to identify both the likelihood and damage potential of water hammer events. Furthermore, the influence of thermal stratification on the thermal hydraulic behavior of the AP600 also remains to be evaluated.

SBWR Program

The objective of the PUMA test program is to obtain data for assessing computer code simulation of important SBWR-specific phenomena. The focus of this test program is on the operability of the passive cooling systems and their interactions with the reactor vessel.

Again, a PIRT has not been completed. The PIRT effort should be brought to a close so that a proper evaluation of PUMA and the GENE test facilities (GIST, GIRAFFE, and PANDA) can be made.

Scaling of phenomena identified in the Purdue University preliminary PIRT has been a major part of the PUMA test program. At present, the scaling effort has primarily focused on the details of local phenomena whereas global scaling appears to be incomplete. To preclude atypicalities in the interactions of the various systems and to help determine an appropriate set of initial and operating conditions for the PUMA system, the scaling of the global dynamic component interactions (among the reactor vessel, drywell, wetwell, PCCS, ICS, and GDCS) should be completed before the facility design is frozen.

We are pleased to see that one of the PUMA program principal investigators is a code developer. Input from a code developer on the selection of instrument type, number, and location will yield a much more useful set of data for code assessment.

The PUMA facility will allow testing that both overlaps and extends the accident period covered by the GENE test facilities (GIST, GIRAFFE, and PANDA), while allowing the simulation of a broad spectrum of postulated accidents. This should be helpful in confirming the validity of the results obtained at the GENE facilities.

The following comments are specific to the PUMA program:

- The current plan is to measure the heat transfer characteristics and infer the noncondensible gas concentration. We would like to point out that knowledge of the noncondensible gas distribution is fundamental and necessary if one is to avoid compensating errors in the computational process. We recommend that the noncondensible gas concentration be measured directly at several locations.

- The test matrix does not include a long-duration test. We believe it should because the SBWR containment performance requirement is 72 hours, which scales to 144 hours of PUMA test time.

- Since the interface temperature of the suppression pool is directly coupled to the containment pressure, an evaluation of thermal stratification in the pool is needed.

- Some tests should be conducted with initial nitrogen concentrations in the drywell to evaluate the impact of steam line breaks outside containment.

- The planning of the PUMA experiments should include consideration of phenomena arising as a consequence of failures of active mitigating systems.

- Data analysis and evaluation are not part of the contract with Purdue University. This is unfortunate because in this case the principal investigators at Purdue University are highly qualified for such a task. Further, those conducting the testing can bring valuable insights to the process. We recommend that the contract with Purdue University be modified to include a data analysis and evaluation task.

Technical Oversight

The RES staff now plans technical oversight of thermal hydraulic research for the AP600 and the SBWR through the Advanced Light Water Reactor Thermal Hydraulic Research Integration Group (ATRIG). This unwieldy ATRIG is not the technical oversight recommended by the ACRS in the past and subsequently approved by the Commission. Lessons learned from the CSAU program should be remembered. A small (5 or 6 members) cohesive group with well-qualified leadership is needed to integrate the technical issues of scaling, data collection, data analysis, and code development.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

References:
1. "Summary of the LSTF Characterization Tests Performed in Conjunction with the ROSA/AP600 Experiments," R. A. Shaw, et al., Draft report dated August 1, 1994, transmitted by memorandum dated August 5, 1994, from G. S. Rhee, Office of Nuclear Regulatory Research
2. U. S. Nuclear Regulatory Commission, Draft NUREG/CR, PU-NE 94/1, Subject: Scientific Design of Purdue University Multi-dimensional Integral Test Assembly (PUMA) for GE SBWR, July 1994, transmitted by memorandum dated August 4, 1994, from J. T. Han, Office of Nuclear Regulatory Research
3. Memorandum dated August 8, 1994, from M. Ishii, Purdue University, to J. Han, U. S. Nuclear Regulatory Commission, transmitting replacement pages for the report, "Preliminary Scientific Design of Purdue University Multi-dimensional Integral Test Assembly (PUMA) for GE SBWR"
4. U. S. Nuclear Regulatory Commission, NUREG/CR-6066, EGG-2705, "Scaling and Design of LSTF Modifications for AP600 Testing," T. J. Boucher, et al., August 1994
5. SECY-94-138, memorandum dated May 20, 1994, from James M. Taylor, Executive Director for Operations, NRC, for the Commissioners, Subject: Confirmatory High Pressure Integral System Testing of the Westinghouse AP600 Safety Systems
6. "Quick Look Report for ROSA/AP600 Experiment AP-CL-03," R. A. Shaw, et al., undated rough draft, transmitted by memorandum dated August 5, 1994, from G. S. Rhee, Office of Nuclear Regulatory Research
7. Advisory Committee on Reactor Safeguards Report, dated November 18, 1993, from J. Ernest Wilkins, Jr., ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: NRC Confirmatory Test Program in Support of the AP600 Design Certification