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Volume 17

A Compilation of Reports of The Advisory Committee on Reactor Safeguards

1995 Annual

U.S. Nuclear Regulatory Commission

April 1996

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ABSTRACT

This compilation contains 44 ACRS reports submitted to the Commission, or to the Executive Director for Operations, during calendar year 1995. It also includes a report to the Congress on the NRC Safety Research Program. All reports have been made available to the public through the NRC Public Document Room and the U. S. Library of Congress. The reports are divided into two groups: Part 1: ACRS Reports on Project Reviews, and Part 2: ACRS Reports on Generic Subjects. Part 1 contains ACRS reports by project name and by chronological order within project name. Part 2 categorizes the reports by the most appropriate generic subject area and by chronological order within subject area.

PREFACE

The enclosed reports represent the recommendations and comments of the U. S. Nuclear Regulatory Commission's Advisory Committee on Reactor Safeguards during calendar year 1995. NUREG-1125 is published annually. Previous issues are as follows:

Volume	Inclusive Dates
1 through 6	September 1957 through December 1984
7	Calendar Year 1985
8	Calendar Year 1986
9	Calendar Year 1987
10	Calendar Year 1988
11	Calendar Year 1989
12	Calendar Year 1990
13	Calendar Year 1991
14	Calendar Year 1992
15	Calendar Year 1993
16	Calendar Year 1994

<u>ACRS MEMBERSHIP (1995)</u>

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	Dr. Dana A. Powers Sandia National Laboratories
	Dr. William J. Shack Argonne National Laboratory
	Mr. Charles J. Wylie, Retired Duke Power Company

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Part 1: ACRS Reports on Project Reviews

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

November 8, 1995

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

Dear Chairman Jackson:

SUBJECT: APPLICATION FOR OPERATING LICENSE FOR WATTS BAR NUCLEAR PLANT UNIT 1

During the 426th meeting of the Advisory Committee on Reactor Safeguards, November 2-4, 1995, we reviewed the application of the Tennessee Valley Authority (TVA) for a license to operate the Watts Bar Nuclear Plant Unit 1. The Watts Bar Subcommittee also discussed this matter at a meeting on November 1, 1995. During the meetings, we had the benefit of discussions with representatives of the NRC staff and the TVA staff, and several members of the public. We also had the benefit of the documents referenced. Several ACRS members visited the site on October 3, 1995. The Committee previously reported on the TVA application on August 16, 1982.

Watts Bar Nuclear Plant Unit 1 is located in eastern Tennessee. The unit employs a Westinghouse nuclear steam supply system with a rated core power level of 3411 MWt and has an ice-condenser containment. The design is similar to that of the Sequoyah Nuclear Plant Units 1 and 2, which received their operating licenses in September 1980 and September 1981, respectively.

In its August 16, 1982 report, the Committee concluded that the Watts Bar units could be operated without undue risk to the health and safety of the public subject to the satisfactory completion of construction, staffing, and preoperational testing, as well as to the resolution of the following concerns: a serious quality assurance breakdown, flow-induced vibration in the steam generators, the integrity of the cement lining of the essential raw cooling water system piping, and the acceptability of the hydrogen control system.

There has been a long history of construction quality problems leading to a number of work stoppages at Watts Bar. With the restart of construction in December 1991, TVA's corrective actions have resulted in improvements in its quality assurance program. The staff has concluded that current performance indicates that TVA has overcome significant weaknesses identified in the past and that TVA's recent performance is satisfactory. Plant construction is now essentially complete and TVA has conducted a successful hot functional test.

We discussed the status of the concerns noted above during our 415th meeting of November 3-4, 1994, and our 426th meeting of November 2-4, 1995. We believe that TVA and the staff have adequately addressed these concerns. During our discussions, the Watts Bar management expressed its commitment to operational excellence and to establishing an effective safety culture. It is our view that TVA's commitment is genuine, but that achieving and maintaining an effective safety culture will require continued senior management involvement.

The NRC staff stated, in Supplement 18 to the Watts Bar Safety Evaluation Report, that all licensing issues have been resolved with the exception of those related to fire barrier penetration seals and emergency lighting inside the reactor building. As a result of our review, we have not identified any new safety concerns.

We believe that, subject to resolution of the open issues to the satisfaction of the staff, there is reasonable assurance that Watts Bar Nuclear Plant Unit 1 can be operated at core power levels up to 3411 MWt without undue risk to the health and safety of the public.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

References:
1. U. S. Nuclear Regulatory Commission, NUREG-0847, "Safety Evaluation Report Related to the Operation of Watts Bar Nuclear Plant, Units 1 and 2," through Supplement 18, issued October 1995
2. U. S. Nuclear Regulatory Commission, NUREG-1528, "Reconstitution of the Manual Chapter 2512 Construction Inspection Program for Watts Bar Unit 1," issued September 1995
3. Letter dated August 16, 1982, from Paul Shewmon, ACRS Chairman, to Nunzio J. Palladino, NRC Chairman, Subject: ACRS Report on Watts Bar Nuclear Plant, Units 1 and 2
4. Letter dated October 26, 1995, from Paul Gunter, Nuclear Information and Resource Service, to Noel Dudley, ACRS, Subject: Public Concerns With Fire Protection Issues At Watts Bar Nuclear Power Station

5. Additional documents submitted to the Committee by members of the public at ACRS meetings November 1-2, 1995

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

November 8, 1995

MEMORANDUM TO: James M. Taylor
Executive Director for Operations

FROM: John T. Larkins, Executive Director
Advisory Committee on Reactor Safeguards

SUBJECT: ALLEGATIONS CONCERNING THE APPLICATION FOR OPERATING LICENSE FOR WATTS BAR NUCLEAR PLANT UNIT 1

During the 426th meeting of the Advisory Committee on Reactor Safeguards, November 2-4, 1995, and the meeting of the Watts Bar Subcommittee, November 1, 1995, we heard presentations by members of the public concerning the Tennessee Valley Authority (TVA) application for an operating license for Watts Bar Nuclear Plant Unit 1. During the meetings, members of the public raised several allegations related to activities involving NRC and TVA personnel. The Committee was uncertain whether these allegations had been previously received and resolved by the agency, and decided that the information should be turned over to the appropriate NRC organizations.

Based on communications with Region II, we have sent the portion of the meeting transcripts that contain the public comments and the documents provided to the ACRS by members of the public to the Office of the Inspector General, the Office of Investigations, and Region II. The NRC Watts Bar Allegations Coordinator plans to review the transcripts and documents for any unidentified safety issues.

cc: J. Hoyle, SECY
L. Soffer, OEDO

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Part 2: ACRS Reports on Generic Subjects

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

January 19, 1995

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

Dear Mr. Taylor:

SUBJECT: GENERAL ELECTRIC NUCLEAR ENERGY TEST AND ANALYSIS PROGRAM FOR THE SIMPLIFIED BOILING WATER REACTOR DESIGN

During the 417th meeting of the Advisory Committee on Reactor Safeguards, January 12-13, 1995, we reviewed the General Electric Nuclear Energy (GENE) Test and Analysis Program (TAP) being conducted in support of the Simplified Boiling Water Reactor (SBWR) design certification. Our Subcommittee on Thermal Hydraulic Phenomena reviewed issues associated with this matter at meetings held on August 24, 1994, December 15-16, 1994, and January 10, 1995. During this review, we had the benefit of discussions with representatives of the NRC staff and GENE. We also had the benefit of the documents referenced.

GENE has described the TAP in a report titled, "SBWR Test and Analysis Program Description." This report provides a comprehensive integrated plan for developing the analytical tools needed to analyze the thermal hydraulic performance of the SBWR design. The TAP encompasses the technical requirements for analyzing transients, ATWS, LOCA/ECCS, core power stability, and the Passive Containment Cooling System (PCCS). The GENE TAP represents a significant step forward and a sincere effort to undertake a meaningful program. The program plan, however, lacks sufficient detail for us to conclude that the various test programs are adequate.

In response to a request from GENE, the NRC staff prepared a draft Safety Evaluation Report (SER) that addresses the adequacy of the TAP. Although we are in basic agreement with positions stated in the SER, we would like to amplify some issues.

An important step that must be taken early in the development of a reliable thermal hydraulic predictive tool is to determine which portions of the modeling require confirmation. GENE does not appear to have completed this step for the development of its TRACG code. The next step is to obtain the needed data from properly scaled separate effects and integral facilities. The scaling analysis of the facilities should be a front-end item. For this program, it was not. As a result, the adequacy of the data obtained from the test facilities employed by GENE will be in question. We recommend that GENE address this scaling issue soon to resolve questions concerning data adequacy and to avoid the possibility of a major delay in the SBWR design certification. Specifically, we recommend that GENE undertake the following tasks: perform a global scaling analysis, identify the important scaling groups and their respective ranges, identify distortions of the plant systems resulting from compromises in test configurations, and describe the models that have been, or will be, added to TRACG for evaluation of important phenomena.

The simulation of the thermal hydraulic phenomena in the PCCS and containment (drywell and wetwell) under accident conditions represents a new challenge to the TRACG code. The code must be shown to be capable of predicting the distributions of noncondensible gases (nitrogen and hydrogen) within the containment volumes before the effectiveness of the PCCS can be demonstrated to a known certainty. Validation of this modeling requires that data on the distribution of noncondensible gases be obtained from a suitably scaled facility. Two candidate facilities (PANDA and GIRAFFE) exist, but we do not believe that the current instrumentation in either is adequate to obtain the needed data. The data required for code validation should be determined, and the test facilities and their instrumentation should be modified accordingly.

The focus of the present GENE test program is on evaluation of long-term core cooling. Previous studies of the early blowdown phase are assumed to be applicable to the SBWR. The intermediate period, sometimes called the Gravity-Driven Cooling System period, is not properly addressed. The staff believes additional testing is needed to obtain data for this period. We agree with this view and encourage GENE to perform the needed testing.

We understand that GENE plans to resolve the issues identified by the staff as well as those identified by us. We wish to review the revised version of the TAP document.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

References:
1. Memorandum dated November 25, 1994, from Dennis Crutchfield, Office of Nuclear Reactor Regulation, to Executive Director, ACRS, transmitting draft Safety Evaluation Report on the Adequacy of the Technical Approach to the Testing and Analysis Program for the Simplified Boiling Water Reactor Design

2. GE Nuclear Energy Topical Report, NEDC-32391P, Revision A, "SBWR Test and Analysis Program Description," September 1994 (Proprietary Document).
3. Letter dated March 7, 1994, from Dennis Crutchfield, Office of Nuclear Reactor Regulation, to Patrick Marriott, GENE, Subject: Simplified Boiling Water Reactor (SBWR) Testing Program
4. GE Nuclear Energy Topical Report, NEDE-32177P, Revision 1, "TRACG Qualification," June 1993 (Proprietary Document)
5. GE Nuclear Energy Licensing Topical Report, NEDE-32176P, "TRACG Model Description," February 1993 (Proprietary Document)
6. GE Nuclear Energy Topical Report, NEDE-32178P, Revision 0, "Application of TRACG Model to SBWR Licensing Safety Analysis." February 1993 (Proprietary Document)

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

April 12, 1995

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

Dear Mr. Taylor:

SUBJECT: NRC TEST AND ANALYSIS PROGRAM IN SUPPORT OF AP600 ADVANCED LIGHT WATER PASSIVE PLANT DESIGN REVIEW

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

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

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

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

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

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

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

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

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

Sincerely,

T. S. Kress

T. S. Kress
Chairman

References:

1. Memorandum dated February 14, 1995, from M. Wayne Hodges, NRC Office of Nuclear Regulatory Research, to John T. Larkins, ACRS Executive Director, transmitting INEL draft report, "Interim Phenomena Identification and

Ranking Tables for Westinghouse AP600 Small Break Loss-of-Coolant Accident, Main Steam Line Break, and Steam Generator Tube Rupture Scenarios," INEL-94/0061

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

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

June 15, 1995

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

Dear Mr. Taylor:

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

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

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

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

I. Leak-Before-Break Approach

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

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

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

II. Security Design

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

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

III. Technical Specifications

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

IV. Initial Test Program

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

V. Passive System Thermal-Hydraulic Performance Reliability

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

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

VI. Regulatory Treatment of Non-Safety Systems

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

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

VII. Containment Performance

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

VIII. External Reactor Vessel Cooling

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

IX. Passive Hydrogen Control Measures

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

X. DBA and Long-Term Severe Accident Radiological Consequences

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

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

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

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

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

Sincerely,

T. S. Kress

T. S. Kress
Chairman

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

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

February 15, 1995

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

Dear Mr. Taylor:

SUBJECT: REACTOR WATER CLEANUP SYSTEM LINE BREAK FOR OPERATING BWRS

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

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

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

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

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

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

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

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

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

Sincerely,

T. S. Kress

T. S. Kress
Chairman

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

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

May 17, 1995

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

Dear Mr. Taylor:

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

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

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

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

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

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

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

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

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

We recommend prompt attention to these matters.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

References:

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

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

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

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

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

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

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

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

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

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

November 14, 1995

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

Dear Chairman Jackson:

SUBJECT: PROPOSED MODIFICATIONS TO THE BOILING WATER REACTOR OWNERS GROUP EMERGENCY PROCEDURE GUIDELINES TO ADDRESS REACTOR CORE INSTABILITIES

During the 426th meeting, November 2-4, 1995, the Advisory Committee on Reactor Safeguards completed its review of the proposed modifications to the Boiling Water Reactor Owners Group (BWROG) emergency procedure guidelines (EPGs) to address mitigation of reactor core instabilities. We previously considered this matter during our January and November 1994, and February 1995 meetings. Our Subcommittee on Thermal Hydraulic Phenomena met to consider this matter on May 12, 1993, and January 27 and October 31, 1995. We also had the benefit of the referenced documents.

The ACRS was requested to review BWR core power stability shortly after the March 1988 instability event that occurred at the LaSalle County Station, Unit 2. In the June 14, 1989 report on this matter, we stated that the program developed by the BWROG in conjunction with General Electric Nuclear Energy (GENE) adequately responded to the issue, provided that the reactor protection system functions on demand. We noted, however, that additional study was needed to address safety issues resulting from an anticipated transient without scram (ATWS) event that may be compounded by core power instability.

The BWROG strategy to deal with an ATWS event with core power oscillations is to change the EPGs to instruct the operator to immediately lower the vessel water level below the feedwater sparger. This measure effectively mitigates the consequences of large amplitude core power oscillations. Additionally, the BWROG has recommended that the operators be instructed to lower the vessel water level below the top of active fuel to reduce core power further while liquid boron is being injected into the lower plenum of the vessel. After the required amount of boron has been added, the water level is then to be raised to reinitiate natural circulation and remix the stratified boron mixture to shut down the plant.

The BWROG has not presented a convincing argument that the remixing of the stratified boron would occur in a timely manner. The data presented to support its arguments were derived from 1/6-scale tests performed by GENE nearly 15 years ago, and information has been lost that could support a convincing case for rapid remixing. Moreover, the boron remixing tests were an afterthought, added to a program the primary purpose of which was to provide a more robust evaluation of boron mixing. As a result, some aspects of the facility scaling were deficient. Other relevant data obtained from BWR plant transients do not necessarily support the remixing results from the GENE 1/6-scale tests.

Moreover, calculations by the NRC staff show that maintaining the vessel water level at five feet above top of active fuel is the preferable strategy when standby liquid control (SLC) injection is available. For the lower probability case where SLC injection is unavailable, lowering the vessel water level into the core region in accordance with the BWROG strategy may have a small advantage in gaining time for the operator to take action to restore SLC injection before the suppression pool reaches its temperature limit. Based on calculations made with various computer codes, the BWROG has estimated this time increase to be on the order of 6 to 12 minutes, depending on the assumptions made about plant operating parameters. The NRC staff, however, used the TRAC-B and RAMONA codes to estimate this time increase to be from 1 to 6 minutes. We have greater confidence in the results based on the NRC codes.

Because of the very low probability of an ATWS event, the Backfit Rule precludes requiring the BWROG to revise its previously approved vessel water level control strategy. We therefore concur with the NRC staff position that allows a licensee the option of using the water level control strategy advocated by the BWROG. We strongly urge, however, that the BWR plant licensees reconsider their position in this matter.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

References:

1. Memorandum dated October 12, 1995, from R. Jones, Nuclear Regulatory Commission, to P. Boehnert, ACRS, transmitting draft SER on the Acceptance of the BWROG Emergency Procedure Guidelines Modifications to Address Reactor Core Instabilities

2. Letter dated September 15, 1995, from R. Pinelli, Chairman, BWR Owners Group, to G. Holahan, Nuclear Regulatory Commission, Subject: Request for Comment on Draft Safety Evaluation of Proposed Emergency Procedure Guidelines - Boiling Water Reactor Owners Group (BWROG) Response and attachment, General Electric Report, GE-NE-A00-05652-03 (Proprietary), "Summary of BWR Boron Mixing," dated September 1995
3. SECY-95-002, Memorandum dated January 3, 1995, from James M. Taylor, NRC Executive Director for Operations, for the Commissioners, Subject: Status of the Review of Modifications to Emergency Procedure Guidelines for Boiling Water Reactor ATWS with Power Oscillations
4. Memorandum dated November 4, 1994, from J. March-Leuba, Oak Ridge National Laboratory, to L. Phillips, Nuclear Regulatory Commission, Subject: Estimation of Density Reactivity Coefficient in the Presence of Boron
5. Letter dated January 18, 1995, from J. Dale, GE Nuclear Energy, to U. S. Nuclear Regulatory Commission, Attention P. Boehnert, ACRS, transmitting:

   - NEDC-22030, Boron Remixing Tests (partial copy) - Proprietary
   - NEDE-22267, Test Report, Three-Dimensional Boron Mixing Model (partial copy) - Proprietary
   - NEDE-22275, Evaluation of 1/6-Scale, Three-Dimensional Simulated Boron Mixing Test Results (partial copy) - Proprietary
   - Software Functional Specification for the 3-D Mixing Test DAS - Proprietary
   - BWR/5-218, 1/6-Scale RPV 3-D Model Drawings - Proprietary
   - Sample Calculation of Chemical Composition from Specific Gravity - Proprietary
   - NEDO-20748, Startup Test Program (Japanese Plant)
   - "Volatility of Sodium Pentaborate," an Abstract

6. Report dated June 14, 1989, from Forrest J. Remick, ACRS Chairman, to Lando W. Zech, Jr., NRC Chairman, Subject: Boiling Water Reactor Core Power Stability

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

May 11, 1995

MEMORANDUM TO: James M. Taylor
Executive Director for Operations

FROM: John T. Larkins, Executive Director
Advisory Committee on Reactor Safeguards

SUBJECT: PROPOSED SUPPLEMENT 2 TO NUREG-0654 CONCERNING CRITERIA FOR EMERGENCY PLANNING IN AN EARLY SITE PERMIT APPLICATION

During the 421st meeting of the Advisory Committee on Reactor Safeguards, May 4-6, 1995, the Committee decided not to review the proposed Supplement 2 to NUREG-0654.

Reference:
SECY-95-090 dated April 11, 1995, from James Taylor, Executive Director for Operations, to the Commissioners, Subject: Emergency Planning Under 10 CFR Part 52, with Supplement 2 to NUREG-0654 Concerning Criteria for Emergency Planning in an Early Site Permit Application

cc: J. Hoyle, SECY
J. Blaha, OEDO
M. Taylor, OEDO
L. Spessard, NRR
C. Miller, NRR
F. Kantor, NRR
G. Sege, RES

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

May 10, 1995

MEMORANDUM TO: James M. Taylor
Executive Director for Operations

FROM: John T. Larkins, Executive Director
Advisory Committee on Reactor Safeguards

SUBJECT: PROPOSED NRC GENERIC LETTER REGARDING INADEQUATE TESTING OF SAFETY-RELATED LOGIC CIRCUITS

During the 421st meeting of the Advisory Committee on Reactor Safeguards, May 4-6, 1995, the Committee decided to review the subject generic letter after the public comments have been reconciled by the staff. The Committee has no objection to the staff proposal to issue this proposed generic letter for public comment.

Reference:
Memorandum dated April 21, 1995, from B. K. Grimes, NRR, to J. T. Larkins, ACRS, Subject: Forwarding of Proposed NRC Generic Letter Regarding Inadequate Testing of Safety-Related Logic Circuits

cc: J. Hoyle, SECY
J. Blaha, OEDO
M. Taylor, OEDO
B. Grimes, NRR
A. Chaffee, NRR
A. Kugler, NRR
G. Sege, RES

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

May 15, 1995

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

Dear Mr. Taylor:

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

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

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

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

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

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

Sincerely,

T. S. Kress

T. S. Kress
Chairman

References:

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

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

July 20, 1995

MEMORANDUM TO: James M. Taylor
Executive Director for Operations

FROM: John T. Larkins, Executive Director
Advisory Committee on Reactor Safeguards

SUBJECT: PROPOSED NRC GENERIC LETTER 95-XX TITLED "PRESSURE LOCKING AND THERMAL BINDING OF SAFETY-RELATED POWER-OPERATED GATE VALVES"

During the 423rd meeting of the Advisory Committee on Reactor Safeguards, July 13-14, 1995, the Committee decided not to review the subject generic letter.

cc: J. Hoyle, SECY
J. Blaha, OEDO
M. Taylor, OEDO
B. Sheron, NRR
R. Wessman, NRR
W. Reckley, NRR
G. Sege, RES

Reference:
Memorandum dated June 28, 1995, from F. Miraglia, NRR, to E. Jordan, CRGR, transmitting Proposed Generic Letter 95-XX on Pressure Locking and Thermal Binding of Safety-Related Power Operated Gate Valves

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

July 20, 1995

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

Dear Mr. Taylor:

SUBJECT: PROPOSED PRIORITY RANKINGS OF GENERIC ISSUES: NINTH GROUP

During the 422nd meeting of the Advisory Committee on Reactor Safeguards, June 8-10, 1995, we reviewed the priority rankings proposed by the NRC staff for the generic issues listed in the attached Table A. During this meeting, we had the benefit of discussions with representatives of the NRC staff. We also discussed this matter during our 423rd meeting on July 13-14, 1995.

Our comments on various generic issues considered during this meeting are contained in the following attachments:

Attachment 1 lists those generic issues for which we agree with the proposed priority rankings.

Attachment 2 identifies the issues for which we agree with the proposed priority rankings, but have comments.

Attachment 3 identifies the issue for which we disagree with the proposed priority ranking.

In addition to our comments on the priority rankings of those issues considered at this time, we are concerned that the prioritization process is not timely for some generic safety issues. Currently, three identified issues still await assignment of priority. One was first identified for prioritization in February 1991. However, we note that for the issues scheduled for resolution, the timeliness of resolution appears to be improving.

We note that often the title of a generic issue is much broader than the scope of the issue actually being addressed in the determination of priority. Examples are GSI-149, "Adequacy of Fire Barriers," and GSI-160, "Spurious Actuations of Instrumentation Upon Restoration of Power." Although we may agree with the priority assigned to the narrow issue defined by the scope, we do not wish to imply that we would agree that such a priority is necessarily appropriate for the larger issue denoted by the title.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

Attachments:
As noted above

TABLE A

GENERIC ISSUES REVIEWED BY THE ACRS DURING THE 422ND MEETING, JUNE 8-10, 1995

Generic Issue Number	Title	Priority Ranking Proposed by the NRC Staff	Reference Document
149	Adequacy of Fire Barriers	LOW	Memorandum from E. Beckjord to W. Minners, Oct 19, 1992
158	Performance of Safety-Related Power-Operated Valves Under Design-Basis Conditions	MEDIUM	Memorandum from E. Beckjord to J. Murphy, Jan 26, 1994
159	Qualification of Safety-Related Pumps While Running on Minimum Flow	DROP	Memorandum from E. Beckjord to W. Minners, Sep 22, 1993
160	Spurious Actuations of Instrumentation Upon Restoration of Power	DROP	Memorandum from E. Beckjord to W. Minners, Sep 30, 1993
161	Use of Non-Safety-Related Power Supplies in Safety-Related Circuits (previously called "Associated Circuits")	DROP	Memorandum from E. Beckjord to T. Murley, Mar 12, 1993
162	Inadequate Technical Specifications for Shared Systems at Multiplant Sites When One Unit is Shut Down	DROP	Memorandum from E. Beckjord to W. Minners, Jul 29, 1993
164	Neutron Fluence in Reactor Vessel	DROP (Ongoing RES efforts adequately address this issue.)	Memorandum from E. Beckjord to T. Murley, Nov 30, 1992
165	Spring-Actuated Safety and Relief Valve Reliability	HIGH	Memorandum from E. Beckjord to W. Minners, Nov 26, 1993

Generic Issue Number	Title	Priority Ranking Proposed by the NRC Staff	Reference Document
166	Adequacy of Fatigue Life of Metal Components	NEARLY RESOLVED	Memorandum from E. Beckjord and T. Murley to J. Sniezek, Apr 1, 1993
167	Hydrogen Storage Facility Separation	LOW	Memorandum from E. Beckjord to J. Murphy, Sep 29, 1994
168	Environmental Qualification of Electrical Equipment	NEARLY RESOLVED	Memorandum from E. Beckjord and T. Murley to J. Sniezek, Apr 1, 1993

ATTACHMENT 1

LIST OF GENERIC ISSUES FOR WHICH THE ACRS AGREES WITH THE PRIORITY RANKINGS PROPOSED BY THE NRC STAFF

Generic Issue No.	Title
158	Performance of Safety-Related Power-Operated Valves Under Design-Basis Conditions
159	Qualification of Safety-Related Pumps While Running on Minimum Flow
161	Use of Non-Safety-Related Power Supplies in Safety-Related Circuits
164	Neutron Fluence in Reactor Vessel
165	Spring-Actuated Safety and Relief Valve Reliability
166	Adequacy of Fatigue Life of Metal Components
167	Hydrogen Storage Facility Separation
168	Environmental Qualification of Electrical Equipment

ATTACHMENT 2

GENERIC ISSUES FOR WHICH THE ACRS AGREES WITH THE PRIORITY RANKINGS PROPOSED BY THE NRC STAFF BUT WITH COMMENTS

Generic Issue No.: 160

Title: Spurious Actuations of Instrumentation Upon Restoration of Power

Proposed Priority Ranking: DROP

ACRS Comment: The scope of the prioritization analysis appears limited to the risk associated with (1) inadvertent actuation of low-temperature overpressure-protection relief valve and (2) inter-system LOCA due to inadvertent opening of a low-pressure safety-injection (LPSI) discharge valve, combined with check valve failure, resulting in over-pressurization of the LPSI system from reactor coolant system pressure. This scope seems overly narrow, particularly in view of the continuing digitization of instrumentation and control (I&C) systems in operating nuclear power plants. The digitization of I&C systems warrants careful reconsideration of issues which originated with analog-based I&C systems, but which may become more risk significant due to the nature of digital technology. It may be appropriate to address this issue in the revision to the NRC Standard Review Plan.

Generic Issue No.: 162

Title: Inadequate Technical Specifications for Shared Systems at Multiplant Sites When One Unit is Shut Down

Proposed Priority Ranking: DROP

ACRS Comment: We note that the prioritization analysis did not encompass the Susquehanna spent fuel pool issue, which partly involved shared cooling systems at a multiplant site and upon which we commented in our letter of December 19, 1994. We believe that reconsideration of the scope of systems included in the prioritization analysis may be needed.

ATTACHMENT 3

GENERIC ISSUE FOR WHICH THE ACRS DISAGREES WITH THE PRIORITY RANKING PROPOSED BY THE NRC STAFF

Generic Issue No.:	149
Title:	Adequacy of Fire Barriers
Proposed Priority Ranking:	LOW
ACRS Recommendation:	MEDIUM
Basis:	The focus of this GSI is on overpressurization of fire barrier seals in room penetrations. Nuclear plant fire barrier qualification is usually based on meeting the ASTM-119 or NFPA-251/252 Standards. As noted in NUREG-0933, testing to these Standards does not always simulate realistic nuclear plant fire conditions. Accounting for the difference between these Standards and realistic conditions is a necessary first step to be taken before assessing the safety significance of specific issues such as this one. Accordingly, we believe that additional work needs to be done on this generic issue. Alternatively, such penetration seal issues as overpressurization could be included in the scope of NRR's current task action plan on fire protection requirements.

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

July 20, 1995

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

Dear Mr. Taylor:

SUBJECT: RESOLUTION OF GENERIC SAFETY ISSUE 83, "CONTROL ROOM HABITABILITY"

During the 423rd meeting of the Advisory Committee on Reactor Safeguards, July 13-14, 1995, we heard presentations from the staff concerning resolution of the subject generic safety issue (GSI). We also had the benefit of the documents referenced.

We have had a long-standing interest in a variety of issues relating to control room habitability. The proposed resolution of this GSI deals with two of these issues, meteorological models and toxic chemicals.

The staff has developed meteorological models and computer software (HABIT) that will permit the staff and licensees to make more realistic estimates of radiological doses and toxic-gas exposures of control room personnel to determine compliance with General Design Criterion 19. The improved meteorological models in HABIT are based on reactor-model wind-tunnel tests and reactor-site tracer studies and will supplant the Murphy/Campe models referenced in Standard Review Plan Section 6.4. This extensive experimental program seems to be a promising basis for resolving meteorological concerns. The computer code, EXTRAN, that treats transport from the source to the control room air intake may not be adequate to deal with the wide variety of circumstances that arise. This is a complex arena for computation and any substantive comment by us would require more review of the meteorological models. We will only pursue this if control room habitability is determined by risk analyses to be an important safety issue.

The proposed resolution of GSI-83 is an example of the difficulty that arises in trying to apply design-basis concepts to resolve what is basically a risk issue. The staff appears to be refining the original "conservative" design-basis accident (DBA) approach by taking some of the conservatisms out of the calculational models. The intent of making these new calculations would be to obtain results that meet the DBA acceptance criteria. The problem with this approach is that the level of conservatism in the original DBA calculation has not been determined, nor has an acceptable level of conservatism been defined. We believe that the appropriate resolution of this GSI would be to determine the acceptable risk. This requires a probabilistic treatment and quantified uncertainty using acceptable calculational tools.

The staff is also revising Regulatory Guide 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release," to incorporate revised limits on toxic chemicals. We find the revised limits difficult to justify. The revisions have greatly increased limits found in Regulatory Guide 1.78. In most cases, the revised values are above the concentration limits considered "immediately dangerous to life and health." The limits have been chosen to assure that operators will have time to don breathing apparatus. Of more interest would be toxic chemical concentration limits that assure that any degradation of operator performance would not produce an unacceptable increase in risk. In evaluating degradation of operator performance, consideration should be given to the effects protective actions (wearing breathing apparatus, isolating the control room, etc.) will have on operator performance.

Finally, we discussed the 1988 survey of control room habitability systems at twelve nuclear power plants (NUREG/CR-4960). This program, which was initiated in response to concerns raised by the Committee, showed that there were many "compliance issues" with these systems. The staff told us that it had under consideration special plant inspections to deal with this situation. We wish to be kept informed of this activity.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

References:

1. Memorandum dated June 6, 1995, from M. Wayne Hodges, Director, Division of Systems Technology, RES, to John T. Larkins, Executive Director, ACRS, Subject: Resolution of Generic Safety Issue 83, "Control Room Habitability"
2. NUREG/CR-6210 dated March 10, 1995, Computer Codes for Evaluation of Control Room Habitability (HABIT)
3. NUREG/CR-4960 dated October 1988, Control Room Habitability Survey of Licensed Commercial Nuclear Power Generating Stations
4. NUREG/CR-5669 dated July 1991, Evaluation of Exposure Limits to Toxic Gases for Nuclear Reactor Control Room Operators

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

July 20, 1995

MEMORANDUM TO: James M. Taylor
Executive Director for Operations

FROM: John T. Larkins, Executive Director
Advisory Committee on Reactor Safeguards

SUBJECT: PROPOSED NRC GENERIC LETTER TITLED "RELOCATION OF SELECTED TECHNICAL SPECIFICATIONS REQUIREMENTS RELATED TO INSTRUMENTATION"

During the 423rd meeting of the Advisory Committee on Reactor Safeguards, July 13-14, 1995, the Committee decided not to review the subject generic letter.

cc: J. Hoyle, SECY
J. Blaha, OEDO
M. Taylor, OEDO
B. Sheron, NRR
W. Reckley, NRR
G. Sege, RES

Reference:
Memorandum dated May 31, 1995, from F. Miraglia, NRR, to E. Jordan, CRGR, Subject: Request for Review and Endorsement of the Proposed NRC Generic Letter Titled "Relocation of Selected Technical Specifications Requirements Related to Instrumentation"

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

September 12, 1995

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

Dear Mr. Taylor:

SUBJECT: PROPOSED RESOLUTION OF GENERIC ISSUE 24, "AUTOMATIC ECCS SWITCHOVER TO RECIRCULATION"

During the 423rd meeting of the Advisory Committee on Reactor Safeguards, July 13-14, 1995, we discussed the proposed resolution of Generic Issue 24. During this meeting, we had the benefit of discussions with representatives of the NRC staff. We also had the benefit of the document referenced. We concur with the resolution proposed by the staff.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

<u>Reference</u>:
Memorandum dated May 8, 1995, from C. Serpan, RES, to J. Larkins, ACRS, Subject: Proposed Resolution of Generic Issue 24, "Automatic ECCS Switchover to Recirculation"

cc: J. Hoyle, SECY
J. Blaha, OEDO
L. Soffer, OEDO
D. Morrison, RES
L. Shao, RES
C. Serpan, RES
J. Cortez, RES
A. Thadani, NRR

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

December 13, 1995

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

Dear Mr. Taylor:

SUBJECT: PROPOSED FINAL GENERIC LETTER, "TESTING OF SAFETY-RELATED LOGIC CIRCUITS"

During the 427th meeting of the Advisory Committee on Reactor Safeguards, December 7-8, 1995, we reviewed the proposed final Generic Letter on Testing of Safety-Related Logic Circuits. This Generic Letter was developed by the staff to address continuing problems with licensee actions to correct deficiencies associated with testing of safety-related logic circuits. During this meeting, 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.

Despite a number of Information Notices describing observed deficiencies in the testing procedures for safety-related logic circuits, deficiencies continue to be reported. This suggested to the staff that licensees had not taken sufficient action to correct such deficiencies.

The proposed Generic Letter would (1) notify licensees about problems with testing of safety-related logic circuits, (2) request that all licensees perform an in-depth review of their surveillance procedures and to modify them as required to ensure compliance with technical specifications, and (3) require that all licensees submit a written response regarding implementation of the requested actions. The NRC staff judges this Generic Letter to be a "compliance backfit" per 10 CFR 50.109, because technical specification requirements for testing safety-related logic circuits have not been met. Consequently, the staff determined that a full backfit analysis was not required.

We agree that there is a need for the NRC staff to address the issues associated with testing of safety-related logic circuits in a comprehensive manner. However, we believe this issue could be more appropriately addressed by existing NRC inspection programs. Additionally, the Nuclear Energy Institute has offered to work with the staff in developing a more effective means of ensuring compliance with requirements for testing of safety-related logic circuits.

Additional comments by ACRS Members Thomas S. Kress and William J. Shack are presented below.

Sincerely,

Robert L. Seale

Robert L. Seale
Vice-Chairman

Additional Comments by ACRS Members Thomas S. Kress and William J. Shack

We believe that the actions required by the proposed final Generic Letter are reasonable and appropriate and that the Generic Letter should be issued.

References:
1. SECY-95-287, Memorandum dated December 4, 1995, from James M. Taylor, NRC Executive Director for Operations, to the Commissioners, Subject: Proposed NRC Generic Letter 95-XX, "Testing of Safety-Related Logic Circuits"
2. Memorandum dated October 18, 1995, from Frank J. Miraglia, Office of Nuclear Reactor Regulation, to Edward L. Jordan, Chairman, Committee to Review Generic Requirements, Subject: Request for Endorsement, Without Formal Review, of Proposed Generic Letter, "Testing of Safety-Related Logic Circuits"

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

November 13, 1995

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

Dear Mr. Taylor:

SUBJECT: NUREG-0700, REVISION 1, "HUMAN-SYSTEM INTERFACE DESIGN REVIEW GUIDELINE"

During the 426th meeting of the Advisory Committee on Reactor Safeguards, November 2-4, 1995, we heard presentations by and held discussions with the NRC staff concerning the subject Design Review Guideline. We also had the benefit of the document referenced.

An outgrowth of the Three Mile Island accident was an NRC requirement that all licensees and applicants for commercial nuclear power plant operating licenses conduct detailed control room design reviews, including reviews of remote shutdown panels, to identify and correct design deficiencies related to human factors. Extensive guidelines published as NUREG-0700, "Guidelines for Control Room Design Reviews," were prepared to support these reviews.

The introduction of computer-based, human-system interface (HSI) technology into nuclear power plants prompted the development of Revision 1 to NUREG-0700. The objective of this document is to provide guidance to the NRC staff for HSI reviews of design submittals or as part of an inspection or other type of regulatory review.

The staff has developed technically defensible principles in Part 1 and a set of guidelines for HSI design reviews in Part 2. However, we are concerned that the detailed HSI design review guidance in Part 2 may discourage the approval of other, equally acceptable alternatives. Furthermore, we are concerned that the guidelines in Part 2 will become de facto regulations.

We plan to continue our review of the overall human factors program.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

Reference:
U. S. Nuclear Regulatory Commission, NUREG-0700, Revision 1, "Human-System Interface Design Review Guideline," dated January 1995

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

July 18, 1995

MEMORANDUM TO: Edward L. Jordan, Director
Office for Analysis and Evaluation
of Operational Data

FROM: John T. Larkins, Executive Director R. Caveo for
Advisory Committee on Reactor Safeguards

SUBJECT: COMMENTS BY INDIVIDUAL MEMBERS AND AN ACRS CONSULTANT

Attached for you consideration are comments by Drs. Apostolakis and Miller, ACRS Members, and some of the comments provided by Dr. Kerr, ACRS Consultant, on the NRC technical training programs in the areas of PRA and digital instrumentation and control systems. Please be advised that these comments represent the views of the individuals mentioned above and do not necessarily represent those of the ACRS full Committee.

Attachment:
As stated

cc: ACRS Members
J. Hoyle, SECY
J. M. Taylor, EDO
J. Blaha, EDO
M. Taylor, EDO

COMMENTS BY ACRS MEMBER DR. APOSTOLAKIS

I am concerned about the lowest knowledge, skills, and abilities (KSA) level in the three general groups of skill levels that the program has adopted, i.e., basic, advanced, and expert practitioner levels. The depth of KSA associated with the basic level is not very clear. At the meeting, the staff said that this level trains the staff primarily on PRA results. Given that a PRA contains numerous assumptions and judgments, I question the wisdom of training at this "basic" level, where my understanding is that these assumptions and judgments are not addressed. Could someone who does not appreciate the limitations of inputs and models really utilize PRA results in an intelligent manner? Basic training in probability and statistics would be much more meaningful with the PRA applications reserved for advanced-level courses.

Too much emphasis is placed on Level 1 PRA methods for reactors. As the NRC moves toward risk-based regulation, I believe that it is both important and urgent to train the staff on the use of probabilistic models in connection with engineering models for various processes. Understanding how uncertainty analysis can complement so-called deterministic models is a prerequisite to creating a culture within which risk-based regulation can become a reality. Issues requiring this approach are numerous, e.g., PRA Levels 2 and 3 for reactors, external-event analyses, and in performance assessments for nuclear waste repositories.

A course on external events is being developed. This includes earthquakes, fires, floods, tornadoes, hurricanes, transportation accidents, and others. I question whether a single course can do a good job covering events whose analyses require working knowledge of very diverse scientific disciplines, such as seismicity, ground motion resulting from a given earthquake, combustion, heat transfer in all its guises, and so on. Furthermore, these events, especially earthquakes and fires, are frequently found by PRAs to contribute significantly to risk. Even in an era of limited resources, these events deserve more attention, perhaps at the expense of the detailed courses that are now being offered on Level 1 PRA methods for reactors.

I find it odd that there is no material covering the basic elements of a performance assessment. This application is rather different from the PRA application of probabilities and attendant uncertainties with which most are familiar.

COMMENTS BY ACRS MEMBER DR. MILLER

The Digital I&C Working Group (DWG) is comprised of ten NRC staff members representing the Technical Training Division, NRR, and regional inspectors and was established in 1994 to develop a digital I&C training program targeted to three groups: region-based inspectors, headquarters I&C personnel, and resident inspectors. It was specifically noted that two members of the DWG are regional inspectors who have had extensive experience with complex digital upgrades completed over the past five years.

The DWG has met three times, since its formation, the most recent meeting being held in April 1995. The result of those meetings has been the development of a plan for a program. This plan has identified a topical outline of the broad areas of required training for each of the three groups and definition of the specific target audiences. The DWG also established priority among the three groups with the regional inspectors having the highest priority. Together these groups represent a relatively small target audience of 15 to 20 persons.

The plan developed by the DWG calls for extensive use of "commercial off the shelf" courses to meet the training requirements of the regional inspectors and the headquarters I&C staff and a Regulatory Perspectives Workshop directed toward improving the inspection processes related to digital I&C issues. I concur with the plan proposed by the DWG. Given the small total number in the target audience, use of commercially available courses is an excellent approach for meeting the generic training requirements. The Regulatory Perspectives workshop will provide needed opportunities for the staff to maintain and update its knowledge and to share experiences between individual regions and headquarters. However, I suggest paticipation by members of industry in part of the workshop.

I believe there are several omissions in the list of topics developed by the DWG, most notably Electromagnetic and Radio Frequency Interference (EMI/RFI). Although the ACRS has in the past expressed concern regarding the staff's emphasis on this environmental stressor at the expense of other stressors, I found this omission surprising. I don't believe EMI/RFI should be considered as a major issue, but neither should be forgotten. Other less obvious omissions in the topical list will or have been discussed with members of the DWG on an individual basis.

My review of the course topics proposed by the DWG also raised questions of relevance to digital I&C since most appear to be directed to background or prerequisite topics. Review of a current catalog of courses offered by one commercial company reveals availability of more appropriate material. I, therefore, recommend the DWG complete a thorough review of currently available commercial courseware. I expect to provide further comments on this issue within the next few weeks.

In summary, I concur with the general approach proposed by the DWG but I am concerned by its execution and the rate of progress made to date. With the recent Generic Letter (GL-96-02) and other activities, the industry is positioned to make rapid progress in updating nuclear plant I&C, the regulatory process thus has an opportunity to play a leadership role, but only if there are capable and well-trained staff able to make competent reviews. I plan to contact members of the TTC staff to discuss ideas related to curriculum development.

REPORT BY DR. KERR, ACRS CONSULTANT, ON MEETING OF THE ACRS SUBCOMMITTEE ON TECHNICAL TRAINING PROGRAMS

July 12, 1995

On July 12, 1995, I attended a meeting of the ACRS Subcommittee on Technical Training Programs. The meeting was held to continue a review of training programs that are being organized by the Office for Analysis and Evaluation of Operational Data (AEOD). One of these is concentrating on Probabilistic Risk Assessment (PRA) with the objective of providing knowledge and skills needed to implement the NRC's goal of moving toward a more nearly risk based regulatory system. Of particular interest is the implementation of the Maintenance Rule. The other is aimed to provide knowledge and skills required to inspect and regulate plant systems that use digitally based control.

Generally, the presentations were well organized and informative, and questions were dealt with effectively. Throughout the discussion of plans for future activities it was clear that significant uncertainty exists both because of the absence of a well-developed policy on how to deal with these areas, and because of uncertainties in the financial resources that will be available to the agency. These two sources of uncertainty are probably related, but of the two the lack of policy is more nearly under the control of the NRC. It will continue to hamper the development of appropriate training until it is resolved.

I have the following comments:

1) The material presented in the PRA-related courses appears to be appropriate, and those involved in the presentations, based on vita previously provided to the Subcommittee, are well qualified as practitioners. However the material is extensive, much of it is probably new to many of those taking the courses, and it is packed into a very short period of time. Thus, at best, the course can be expected to provide novices with an idea of what the field is about. Before it can be used with discretion, considerable additional effort on the part of those who have taken the courses will be required. If sufficient enthusiasm is developed by the introduction to this new area, the course material can provide a foundation for further learning. However, additional individual effort will be necessary if this introduction is to be useful. It would be interesting, in the elicitation of student evaluations, if some indication of student enthusiasm for further study could be found.

2) Discussion indicated that although examinations are not yet used at the end of the courses, they will be developed. I recommend especially on the basis of the

large amount of material covered in the course, that these be open book exams.

3) Discussion indicated that at present no organized method exists for getting feedback from the NRC Regional Offices on these courses. It might be useful to have this.

4) The approach used in developing inspection and regulation for digitally based control systems appears to have been ad hoc. Discussion indicated that there is now a relatively well-developed system, but that documentation that gathers together the requirements is not yet complete. This should be developed both for those who do not know where to look for the information that exists, and for those who do. Almost invariably when various disparate sources are mixed and the results are integrated, discrepancies are identified and improvement occurs.

5) At a previous meeting, Dr. Catton had requested an example of forthcoming AEOD reports on the reliability of individual components to demonstrate how the concept of reliability is being used. The document provided was NUREG-1275, Vol 10, "Operating Experience Feedback Report-Reliability of Safety-Related Steam Turbine-Driven Standby Pumps". I looked in vain, in this report, for a demonstration of how the concept of reliability is being used. Further, I looked for some indication of how much influence the observed performance of these systems had on risk. It is unfortunate that after the collection and analysis of all these data, no comments were made on whether the observed performance contributes an unacceptable, or even a significant, level of risk.

William Kerr, Consultant
13 July, 1995

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

September 15, 1995

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

Dear Chairman Jackson:

SUBJECT: DEVELOPMENT OF IMPROVED NONDESTRUCTIVE EXAMINATION (NDE) TECHNIQUES

During the 424th meeting of the Advisory Committee on Reactor Safeguards, September 7-8, 1995, we heard presentations from representatives of the Electric Power Research Institute (EPRI), the EPRI Technical Advisory Group on NDE, Zetec, Babcock & Wilcox Nuclear Technologies, ABB-Combustion Engineering, and Westinghouse Electric Corporation regarding activities to improve NDE techniques for more accurately detecting and assessing steam generator tube defects. The status of staff activities on the development of a new steam generator rule and a supporting research program was also discussed. We had the benefit of the documents referenced.

In the June 16, 1995 Staff Requirements Memorandum, the Commission asked the ACRS to assist the staff in encouraging the industry to develop improved NDE techniques for steam generator tube inspections. The industry presentations at our meeting indicated that substantial progress is being made on the development of techniques that will provide significantly improved capabilities for detecting and sizing circumferential flaws. Not surprisingly, industry efforts are focused on a rapid resolution of the circumferential cracking problem using evolutionary improvements in eddy current technology. In addition, development is proceeding on innovative techniques such as ultrasonic guided (Lamb) waves, in situ fluorescent dye-penetrant inspections, in situ tube burst pressure testing, and combined ultrasonic and eddy current probes. Improved methods of signal processing and display are being developed to aid interpretation of NDE results. We believe modern, real time, signal processing technologies could provide great improvements in signal interpretation, defect detection, and defect sizing.

The staff and industry both recognize that the current regulatory approach to steam generator inspections discourages the development and adoption of improved NDE techniques. In the current framework, an increased detection capability leads to more plugging or repairs without necessarily improving safety. We believe that adoption of a new steam generator rule with realistic requirements for demonstrating tube integrity could provide the industry with a strong economic incentive to develop more effective NDE techniques. Careful thought must be given to the requirements for adequate "performance demonstrations" of the NDE techniques essential for implementing a new rule. The steam generator mockup being developed by Westinghouse Electric Corporation under the Office of Nuclear Regulatory Research sponsorship may provide a useful independent regulatory check on the adequacy of NDE inspection techniques.

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

Sincerely,

T. S. Kress

T. S. Kress
Chairman

References:

1. Staff Requirements Memorandum dated June 16, 1995, from Andrew L. Bates, Acting Secretary of the Commission, Subject: Meeting with ACRS, June 8, 1995
2. NRC Information Notice 94-88, "Inservice Inspection Deficiencies Result in Severely Degraded Steam Generator Tubes," dated December 23, 1994
3. NRC Generic Letter 95-03, "Circumferential Cracking of Steam Generator Tubes," dated April 28, 1995

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

October 13, 1995

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

Dear Chairman Jackson:

SUBJECT: NATIONAL ACADEMY OF SCIENCES/NATIONAL RESEARCH COUNCIL STUDY ON "DIGITAL INSTRUMENTATION AND CONTROL SYSTEMS IN NUCLEAR POWER PLANTS, SAFETY AND RELIABILITY ISSUES" - PHASE 1

During the 425th meeting of the Advisory Committee on Reactor Safeguards, October 5-7, 1995, we reviewed the National Academy of Sciences/National Research Council (NAS/NRC) Phase 1 report on Digital Instrumentation and Control Systems in Nuclear Power Plants, Safety and Reliability Issues. The NAS/NRC Committee Chairman described the results of the Phase 1 report. We also had the benefit of the documents referenced.

The objective of the Phase 1 study was to define the important safety and reliability issues concerning hardware, software, and human-machine interfaces that arise from the use of digital instrumentation and control technology in nuclear power plant operations. The report identifies eight key issues: six technical and two strategic. It notes that these issues are common to other industries where software is required for dependable operation of systems. The report succinctly presents the issues that the NAS/NRC Committee found to be important.

We agree that the issues identified in the Phase 1 report will be important considerations as digital technology is used more extensively in nuclear power plants. In the past, we have called attention to the effects of environmental stressors. The NAS/NRC Chairman stated that the NAS/NRC Committee considered, but decided not to raise this issue to the level of a "key technical issue." We continue to believe this is an important issue that the staff must address as it develops its regulatory guidance for digital systems. However, this is part of the broader issue of environmental qualification of safety-related equipment and does not need to be a key issue of the Phase 2 study.

We have concerns regarding a potential conflict between the Phase 2 completion schedule and the staff's schedule for issuing the Standard Review Plan (SRP) and associated regulatory guides. We believe it is important that the SRP and other regulatory guidance benefit from the insights in the Phase 2 report.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

References:

1. Report dated 1995, from the Committee on Application of Digital Instrumentation and Control Systems to Nuclear Power Plant Operations and Safety, Board on Energy and Environmental Systems, Commission on Engineering and Technical Systems, National Research Council, Subject: Digital Instrumentation and Control Systems in Nuclear Power Plants, Safety and Reliability Issues - Phase 1
2. Memorandum dated December 2, 1993, from Ivan Selin, Chairman, NRC, to NRC Commissioners, Subject: Computers in Nuclear Power Plant Operations
3. Letter dated July 14, 1994, from T. S. Kress, Chairman, ACRS, to Ivan Selin, Chairman, NRC, Subject: Proposed National Academy of Sciences/National Research Council Study and Workshop on Digital Instrumentation and Control Systems
4. Letter dated August 23, 1994, from Ivan Selin, Chairman, NRC, to T. S. Kress, Chairman, ACRS, regarding ACRS letter of July 14, 1994 on National Academy of Sciences/National Research Council Proposal for a Study and Workshop on the "Application of Digital Instrumentation and Control Technology to Nuclear Power Plant Operations and Safety"

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

October 16, 1995

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

Dear Chairman Jackson:

SUBJECT: FATIGUE ACTION PLAN

During the 425th meeting of the Advisory Committee on Reactor Safeguards, October 5-7, 1995, we completed our deliberations on the Fatigue Action Plan that we started during our 424th meeting, September 7-8, 1995. We had the benefit of discussions with representatives of the NRC staff regarding this matter and of the documents referenced.

The Fatigue Action Plan was developed to help resolve Generic Issue 166, "Adequacy of Fatigue Life of Metal Components." It was intended to address three specific issues: (1) the margin against fatigue failure of older nuclear power plants with reactor coolant pressure boundary components designed to ANSI B31.1 requirements rather than the newer ASME Code Section III, Class 1 fatigue requirements; (2) the effects of reactor coolant environments on fatigue life; and (3) the appropriate staff actions when components have cumulative usage factors (CUFs) greater than 1.

The work done on the Fatigue Action Plan by the staff and the additional work supported by the Department of Energy and the Electric Power Research Institute have shown that, even after including environmental effects, the CUFs for almost all reactor components which were originally designed to ASME Code fatigue requirements will still be less than 1. It also showed that the nuclear piping, which had been designed to the ANSI B31.1 requirements, in general has margins against fatigue failure comparable to those achieved by using the ASME Section III, Class 1 fatigue requirements. Although fatigue failures have been experienced in nuclear plants, these failures have been due to unanticipated loads and not to inadequate design margins for the anticipated cyclic loads.

Based on a probabilistic parametric study, the staff concluded that even if fatigue cracks were initiated, rupture of reactor coolant piping as a result of fatigue crack growth would be a low-probability event. We anticipate commenting on this parametric study at a later time.

The summary of the Fatigue Action Plan provides only general guidance for the appropriate actions to be taken when the CUF is greater than 1. However, the supporting documentation suggests that the proposed nonmandatory appendix to Section XI of the ASME Code provides evaluation methods which may be acceptable to the staff. These methods provide a choice of either the traditional CUF approach or a "flaw-tolerance" approach similar to that widely used in the aerospace industry. We agree that these types of evaluations would be appropriate.

We agree with the staff that maintaining the integrity of the reactor coolant pressure boundary is an important element in defense-in-depth, and that fatigue is a potentially significant mechanism which can degrade the integrity of the pressure boundary. But, on the basis of the work done by the staff and industry, no immediate staff or licensee action is needed.

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

Sincerely,

T. S. Kress

T. S. Kress
Chairman

<u>References</u>:

1. Draft Commission Paper, received August 30, 1995, from James M. Taylor, Executive Director for Operations, NRC, to the Commissioners, Subject: Completion of the Fatigue Action Plan (Predecisional)
2. U. S. Nuclear Regulatory Commission, NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components," published March 1995
3. SECY-94-191 dated July 26, 1994, from James M. Taylor, Executive Director for Operations, NRC, to the Commissioners, Subject: Fatigue Design of Metal Components
4. Staff Requirements Memorandum dated May 21, 1993, from Samuel Chilk, Secretary of NRC, to John T. Larkins, Executive Director, ACRS, Subject: Periodic Meeting with the Advisory Committee on Reactor Safeguards, Friday May 14, 1993

5. Letter dated August 17, 1992, from David A. Ward, Chairman, ACRS, to James M. Taylor, Executive Director for Operations, NRC, Subject: Related Branch Technical Position On Fatigue Evaluation Procedures

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

July 20, 1995

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

Dear Mr. Taylor:

SUBJECT: HEALTH EFFECTS VALUATION

During the 423rd meeting of the Advisory Committee on Reactor Safeguards, July 13-14, 1995, we discussed the recent staff reconsideration of the health effects valuation. During this meeting, we had the benefit of discussions with representatives of the staff. We also had the benefit of the document referenced but it differs in some details from the presentation.

In reviewing the health effects valuation, the staff recognized the recent risk coefficients issued by the International Commission on Radiological Protection and retained the linear dose hypothesis. These were used along with the Office of Management and Budget (OMB) recommended value for a "statistical life" to arrive at an indicated increase from the present $1000/person-rem to $2000/person-rem. We were told that such a change is unwarranted because of the order-of-magnitude uncertainty in the regulatory analysis. Consequently, the staff is not proposing to change the value and is considering the following four options for proceeding on this issue:

- Retain the $1000/person-rem but require discounting.

- Retain the $1000/person-rem but require separate quantification of offsite property effects.

- Retain the $1000/person-rem but require both discounting and separate quantification of offsite property consequences.

- Retain status quo in the near term but allow use of the $2000/person-rem subject to discounting and/or separate quantification of offsite property consequences as part of optional sensitivity studies.

We believe that the change in the value is warranted and do not support any of the four options. In the interest of technical correctness, consistency in use across Federal agencies, and regulatory coherence, we recommend use of the new value of $2000/person-rem, as derived from the rounded-off product of the value of a "statistical life" ($3M) and a risk coefficient for the stochastic health effects (7.3×10^{-4} fatalities/person-rem). This value should be used as a dollar proxy for only the health effects associated with dose and should not be used (as in the past with the previous value) as a surrogate for other consequences such as prompt fatalities and land contamination. These other consequences should be evaluated separately as suggested in the draft Federal Register Notice. The MACCS code with an updated economic model would be an appropriate tool for such an evaluation. The new value should be expressed in terms of an identified year's dollars to allow users to make their own correction for inflation. Future effects should be discounted by present worth methods.

The selection of the value of a "statistical life" is the crucial determinant of the value of the health effects conversion factor. We believe that the present most appropriate means of establishing such a value is through the willingness-to-pay approach. This, however, can give a broad range of results that leads to a basic problem of defending the selection of any value from the range. The fact that a value is a median or a mean is not an appropriate defense for its selection in this case. In the absence of knowledge of any rationale underlying the existence of such a broad range, one has little recourse but to fall back on experience and judgment. In this spirit, we propose that there are basically two sound reasons for selecting the value of $3M for a "statistical life".

1. It is specifically cited by the OMB. This is a strong step toward consistency in use across government agencies.

2. Judgment and experience show that it is an appropriate value.

In the past, the $1000/person-rem has been used to represent both exposure and land contamination costs. We believe an exercise should be conducted to develop a sample estimate using the updated MACCS code for the relative magnitude of land contamination costs for severe accidents. Such a comparison would provide guidance on the need for a review of those previous decisions that may have involved predictions of considerable land contamination.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

Reference:
Letter dated March 6, 1995, from Bill M. Morris, Office of Nuclear Regulatory Research, to T.S. Kress, Chairman, ACRS, Transmitting draft Federal Register Notice on Proposed Revision to the Health Effects Valuation. (DRAFT PREDECISIONAL)

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

September 13, 1995

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

Dear Mr. Taylor:

SUBJECT: HEALTH EFFECTS VALUATION

In your September 1, 1995 response to our letter of July 20, 1995, on the referenced subject, you stated that the staff continues to recommend retention of an undiscounted $1000 per person-rem but may also include a discounted $2000 per person-rem as a sensitivity parameter. The justification stated for this recommendation by the staff is that use of the discounted $2000 per person-rem "would misleadingly suggest a level of precision which does not exist," and "would impose additional complications and staff burden, with no improved regulatory decisions."

We continue to recommend the use of a discounted $2000 per person-rem for the following reasons:

(1) Discounting is the technically correct approach for making cost comparisons. An undiscounted single-value surrogate is fundamentally incorrect, and its use can be very misleading. The NRC should not continue to use methods known to be incorrect in its regulatory activities.

(2) The $2000 value embodies an Office of Management and Budget recommended "value of a statistical life." Thus, it is very likely to be adopted by all other U.S. Government agencies in performing mandated cost/benefit analyses for their regulatory activities. There is considerable merit in promoting consistency in methodology across Government agencies.

(3) The $2000 value does not imply a different level of precision than does the $1000 value.

We believe the change to $2000 per person-rem is technically correct and adds to coherency in regulations. We look forward to discussing this matter with the staff in the immediate future.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

References:
1. Letter dated September 1, 1995, frcm James M. Taylor, Executive Director for Operations, NRC, to Thomas S. Kress, Chairman, Advisory Committee on Reactor Safeguards, Subject: Health Effects Valuation
2. Letter dated July 20, 1995, from T. S. Kress, Chairman, Advisory Committee on Reactor Safeguards, to James M. Taylor, Executive Director for Operations, NRC, Subject: Health Effects Valuation

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

June 16, 1995

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

Dear Chairman Selin:

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

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

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

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

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

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

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

Sincerely,

T. S. Kress

T. S. Kress
Chairman

References:
1. SECY-95-126 dated May 18, 1995, from James M. Taylor, Executive Director for Operations, NRC, for the Commissioners, Subject: Final Policy Statement on the Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities (Draft Predecisional)
2. ACRS Report dated May 11, 1994, from T. S. Kress, Chairman, ACRS, to James M. Taylor, Executive Director for Operations, NRC, Subject: Draft Policy Statement on the Use of Probabilistic Risk Assessment Methods in Reactor Regulatory Activities
3. Letter dated January 17, 1995, from William H. Rasin, Nuclear Energy Institute, to Ashok C. Thadani, Office of Nuclear

Reactor Regulation, NRC, transmitting final draft of PSA Applications Guide

4. ACRS Report dated May 13, 1987, from William Kerr, Chairman, ACRS, to The Honorable Lando W. Zech, Chairman, NRC, Subject: ACRS Comments On An Implementation Plan For The Safety Goal Policy

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

February 16, 1995

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

Dear Mr. Taylor:

SUBJECT: PROPOSED FINAL REVISION 3 TO REGULATORY GUIDE 1.118, "PERIODIC TESTING OF ELECTRIC POWER AND PROTECTION SYSTEMS"

During the 418th meeting of the Advisory Committee on Reactor Safeguards, February 9-10, 1995, we reviewed the subject proposed revision to Regulatory Guide 1.118 that provides guidance for implementing some of the requirements of 10 CFR 50.55a(h), "Protection Systems"; 10 CFR Part 50 Appendix A, General Design Criterion (GDC) 18, "Inspection and Testing of Electric Power Systems," and GDC 21, "Protection System Reliability and Testability"; and 10 CFR Part 50 Appendix B, Criterion XI, "Test Control." During our review, we had the benefit of discussions with representatives of the NRC staff and the Institute of Electrical and Electronics Engineers (IEEE). We also had the benefit of the documents referenced.

The proposed revision of Regulatory Guide 1.118 provides updated NRC staff guidance for complying with the Commission's regulations regarding the periodic testing of the electric power and protection systems, and endorses ANSI/IEEE Standard 338-1987, "IEEE Standard Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems," with certain exceptions. The staff intends to use this revision during its evaluation of future applications for construction permits, operating licenses, and licensee modifications to existing nuclear plants that require staff approval.

During the public comment period, the responsible IEEE subcommittee raised the concern that, when a safety system test is initiated by removal of fuses or the opening of a breaker, it may result in undesirable actuation of equipment during plant operations. At our meeting, the NRC staff stated that they were close to a resolution of this concern with IEEE.

Subject to a resolution of the above concern that is acceptable to the staff, we have no objection to the issuance of Regulatory Guide 1.118, Revision 3.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

References:
1. Memorandum dated February 1, 1995, from E. Beckjord, Office of Nuclear Regulatory Research, to J. Larkins, ACRS Executive Director, transmitting Proposed Revision 3 to Regulatory Guide 1.118, "Periodic Testing of Electric Power and Protection Systems"
2. ANSI/IEEE Std 338-1987, "IEEE Standard Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems"

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

April 14, 1995

MEMORANDUM TO: James M. Taylor
Executive Director for Operations

FROM: John T. Larkins, Executive Director
Advisory Committee on Reactor Safeguards

SUBJECT: PROPOSED REVISION 1 TO REGULATORY GUIDE 1.152, "CRITERIA FOR DIGITAL COMPUTERS IN SAFETY SYSTEMS IN NUCLEAR POWER PLANTS"

During the 420th meeting of the Advisory Committee on Reactor Safeguards, April 6-7, 1995, the Committee decided to review the subject revision after the public comments have been reconciled. The Committee does not object to issuing this proposed revision for public comment.

Reference:
Memorandum dated April 4, 1995, from L. C. Shao, RES, to J. T. Larkins, ACRS, Subject: Proposed Revision 1 to the Regulatory Guide 1.152, "Criteria for Digital Computers in Safety Systems of Nuclear Power Plants" (Draft Regulatory Guide DG-1039)

cc: J. Hoyle, SECY
J. Blaha, OEDO
M. Taylor, OEDO
M. Mayfield, RES
S. Aggarwal, RES
G. Sege, RES

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

June 20, 1995

MEMORANDUM TO: James M. Taylor
Executive Director for Operations

FROM: John T. Larkins, Executive Director
Advisory Committee on Reactor Safeguards

SUBJECT: DRAFT REGULATORY GUIDE 1040, "TIME RESPONSE DESIGN CRITERIA FOR SAFETY-RELATED OPERATOR ACTIONS"

During the 422nd meeting of the Advisory Committee on Reactor Safeguards, June 8-10, 1995, the Committee decided to review the subject regulatory guide after the public comments have been reconciled by the staff. The Committee has no objection to the staff proposal to issue this draft regulatory guide for public comment.

Reference:
Draft Memorandum from M. Hodges, NRC Office of Nuclear Regulatory Research, to J. Larkins, ACRS Executive Director, received June 1, 1995, transmitting Draft Regulatory Guide-1040, "Time Response Design Criteria for Safety-Related Operator Actions"

cc: J. Hoyle, SECY
J. Blaha, OEDO
M. Taylor, OEDO
M. Hodges, RES
F. Coffman, Jr., RES
J. Persensky, RES
N. Canfield, RES
J. Kramer, RES
G. Sege, RES

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

July 20, 1995

MEMORANDUM TO: James M. Taylor
Executive Director for Operations

FROM: John T. Larkins, Executive Director
Advisory Committee on Reactor Safeguards

SUBJECT: PROPOSED NRC BULLETIN TITLED "POTENTIAL PLUGGING OF EMERGENCY CORE COOLING SUCTION STRAINERS BY DEBRIS IN BOILING WATER REACTORS" AND PROPOSED REVISION 2 OF REGULATORY GUIDE 1.82 "WATER SOURCES FOR LONG-TERM RECIRCULATION COOLING FOLLOWING A LOSS-OF-COOLANT ACCIDENT"

During the 423rd meeting of the Advisory Committee on Reactor Safeguards, July 13-14, 1995, the Committee decided to review the subject NRC Bulletin and Regulatory Guide revision following the staff's reconciliation of public comments.

cc: J. Hoyle, SECY
J. Blaha, OEDO
M. Taylor, OEDO
R. Elliot, NRR
A. Serkiz, RES
G. Sege, RES

Reference:
Memorandum dated June 9, 1995, from F. Miraglia, NRR, to E. Jordan, CRGR, Subject: Request for Review and Endorsement of the Proposed NRC Bulletin Titled "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling Water Reactors" and Proposed Revision 2 of Regulatory Guide .82 "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

September 13, 1995

MEMORANDUM TO: James M. Taylor
Executive Director for Operations

FROM: John T. Larkins, Executive Director
Advisory Committee on Reactor Safeguards

SUBJECT: PROPOSED REVISION 1 TO REGULATORY GUIDE 1.153, "CRITERIA FOR SAFETY SYSTEMS" (DRAFT REGULATORY GUIDE DG-1042)

During the 424th meeting of the Advisory Committee on Reactor Safeguards, September 7-8, 1995, the Committee decided not to comment on the proposed revision to Regulatory Guide 1.153. The Committee appreciates the staff's efforts to keep it informed of the status of proposed regulatory guide revisions.

Reference:
Memorandum dated August 17, 1995, from Lawrence C. Shao, RES, to John T. Larkins, ACRS, Subject: Proposed Revision 1 Regulatory Guide 1.153, "Criteria for Safety Systems" (Draft Regulatory Guide DG-1042)

cc: J. Hoyle, SECY
J. Blaha, OEDO
L. Soffer, OEDO
S. Aggarwal, RES
J. Cortez, RES
A. Thadani, NRR

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

October 13, 1995

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

Dear Mr. Taylor:

SUBJECT: PROPOSED FINAL REVISION 1 TO REGULATORY GUIDE 1.152, "CRITERIA FOR DIGITAL COMPUTERS IN SAFETY SYSTEMS OF NUCLEAR POWER PLANTS"

During the 425th meeting of the Advisory Committee on Reactor Safeguards, October 5-7, 1995, we reviewed the proposed final Revision 1 to Regulatory Guide 1.152. The revised Regulatory Guide endorses IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations (IEEE Standard 7-4.3.2-1993), "with the exception of quantitative reliability goals (Section 5.15)." During this meeting, we had the benefit of discussions with the NRC staff. We also had the benefit of the documents referenced.

Based on our review, we concur with the Regulatory Position of Revision 1 to Regulatory Guide 1.152. However, we offer the following comment.

In the proposed Regulatory Guide, the staff declines to endorse the use of quantitative reliability goals as the sole means of meeting the Commission regulations for reliability of digital computers in safety systems. This position is consistent with our previously expressed views as provided in our report of March 18, 1993 to Chairman Selin. The language used in the staff response to Public Comment 1 on this issue provides a clearer expression of the staff position on quantitative reliability goals than does the language used in the Regulatory Guide. During our discussion, the staff agreed to modify the language in the Regulatory Guide to be consistent with its response to the public comment.

Subject to the staff's planned modification, we have no objection to the issuance of Regulatory Guide 1.152, Revision 1.

Additional comments by ACRS Members George Apostolakis, Ivan Catton, Mario H. Fontana, William J. Lindblad, and Charles J. Wylie are presented below.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

<u>Additional Comments by ACRS Members George Apostolakis, Ivan Catton, Mario H. Fontana, William J. Lindblad, and Charles J. Wylie</u>

We believe that in taking exception to IEEE 7-4.3.2-1993, Section 5.15, the staff is tilting at windmills. We would endorse the Standard in its entirety. The staff could make its point regarding the adequacy of quantitative reliability goals for software without taking exception to this Section.

<u>References</u>:
1. Regulatory Guide 1.152, Revision 1, dated September 1995, "Criteria for Digital Computers in Safety Systems of Nuclear Power Plants," transmitted by memorandum dated September 1, 1995, from David L. Morrison, NRC Office of Nuclear Regulatory Research, to John T. Larkins, ACRS
2. Institute of Electrical and Electronics Engineers, Standard 7-4.3.2-1993, "Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations," September 15, 1993
3. Letter dated July 31, 1995, from C. L. Terry, Group Vice President, Nuclear, TUELECTRIC, to U.S. NRC, Subject: TU Electric Comments on Draft Regulatory Guide DG-1039, "Criteria for Digital Computers in Safety Systems of Nuclear Power Plants"
4. Report dated March 18, 1993, from Paul Shewmon, Chairman, ACRS, to Ivan Selin, Chairman, NRC, Subject: Computers in Nuclear Power Plant Operations

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

November 14, 1995

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

Dear Mr. Taylor:

SUBJECT: PROPOSED FINAL REGULATORY GUIDE 1.164, "TIME RESPONSE DESIGN CRITERIA FOR SAFETY-RELATED OPERATOR ACTIONS," TO RESOLVE GENERIC SAFETY ISSUE B-17

During the 426th meeting of the Advisory Committee on Reactor Safeguards, November 2-4, 1995, we reviewed the proposed final Regulatory Guide 1.164, which was developed by the staff to resolve Generic Safety Issue B-17, "Criteria for Safety-Related Operator Actions." During the meeting, we had the benefit of discussions with the NRC staff. We also had the benefit of the documents referenced.

Criterion 19, "Control Room," of Appendix A to 10 CFR Part 50, "General Design Criteria for Nuclear Power Plants," requires that a control room be provided from which actions can be taken to operate the nuclear power unit safely under normal and accident conditions. Generic Safety Issue B-17 called for the development of time criteria for safety-related operator actions that included a methodology for determining whether or not automatic actuation would be needed to mitigate a design-basis event.

In Regulatory Guide 1.164, the staff endorses ANSI/ANS-58.8-1994, "Time Response Design Criteria for Safety-Related Operator Actions." This Standard establishes criteria and simplifies the process for calculating the minimum allowable response times for manual operator actions to stabilize the plant during a design-basis event. The NRC staff proposes endorsement of this Standard to resolve Generic Safety Issue B-17.

Based on material presented by the staff, we find no technical basis for the estimates of minimum times for operator actions in ANSI/ANS-58.8-1994. Comparison of the recommended times with results from exercises done on plant simulators does not demonstrate that these times are appropriately conservative. Consequently, we do not support the staff's endorsement of ANSI/ANS-58.8-1994 in Regulatory Guide 1.164 and do not believe that this endorsement is the appropriate way to resolve Generic Safety Issue B-17.

The Standard does not address operator response times for advanced nuclear power plants. There is a need to consider this issue in some way for the evolutionary and passive plants.

Additional comments by ACRS Members George Apostolakis, Ivan Catton, and Robert L. Seale are presented below.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

Additional Comments by ACRS Members George Apostolakis, Ivan Catton, and Robert L. Seale

In support of its recommended minimum response times, the staff relied in part on results that were produced from the Operator Reliability Experiments. We find this to be inappropriate because these experiments were not subjected to independent peer review and the staff did not have access to the actual data collected.

References:

1. Memorandum dated October 4, 1995, from M. Wayne Hodges, Office of Nuclear Regulatory Research, to John T. Larkins, ACRS, Subject: Regulatory Guide 1.164, "Time Response Design Criteria for Safety-Related Operator Actions," for ACRS Review and also transmitting staff response to public comments
2. U. S. Nuclear Regulatory Commission, NUREG-0933, Supplement 06, March 1987, "A Prioritization of Generic Safety Issues," Item B-17, "Criteria for Safety-Related Operator Actions," Revision 2
3. American Nuclear Society, ANSI/ANS-58.8-1994, "Time Response Design Criteria for Safety-Related Operator Actions," approved by the American National Standards Institute, Inc., August 23, 1994

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

May 10, 1995

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

Dear Chairman Selin:

SUBJECT: REGULATORY REFORM INITIATIVES AND NATIONAL PERFORMANCE REVIEW PHASE II

During the 421st meeting of the Advisory Committee on Reactor Safeguards, May 4-6, 1995, we discussed the status of the ongoing Regulatory Reform Initiatives Program (RRIP) and the activities regarding the National Performance Review Phase II (NPR II). During this meeting, we had the benefit of discussions with representatives of NRR, RES, the NRC NPR II Steering Committee, and the Nuclear Energy Institute (NEI). We also had the benefit of the documents referenced. The purpose of our report is to provide comments in a timely manner on the activities of the NPR II Steering Committee.

The NPR II effort draws on the RRIP. The RRIP, which includes elements of the Regulatory Impact Survey (1989), the Regulatory Review Group (RRG) Study (1993), and the RRG Implementation Plan (1994), anticipated the regulatory review aspects of the NPR II requirements. The Cost-Beneficial Licensing Actions and the Requirements Marginal-to-Safety programs demonstrate NRC's commitment to effective and cost-beneficial regulation. As the result of these activities, the NPR II Steering Committee is well positioned to provide specific and detailed recommendations to address the Phase II review of existing regulations

The NPR II also requests a review of the agency mission and an examination of the possible devolution of selected responsibilities to state or local authorities. These issues are being integrated into the Steering Committee recommendations.

The Steering Committee provided us with an outline of the approach to be taken in response to all three areas of concern to the NPR II review. The Steering Committee is tasked to identify burdensome, outdated, marginal-to-safety, overly prescriptive, and overlapping regulations, and to recommend appropriate changes. A review of the functions of the NRC and the efficiency of their implementation will be included.

In response to the request by the Steering Committee, we offer the following comments on its proposed program:

- Those rules and regulations that rely on input from other agencies (such as EPA, NCRP, DOE, DOD, DOS, and DOT) should be identified for future reconciliation with any changes that may arise from those agencies. An obvious example is the NRC interaction with EPA and NCRP on 10 CFR Part 20.
- The Steering Committee report should make it clear that the NRC had launched its intensive review of regulations well before the beginning of NPR II.
- As NRC scrutinizes its regulations, it is imperative that criteria be established for the tradeoff between the requirements of the NRC public health and safety mandate and the goals of the NPR II.

The NEI presented a compilation of proposed changes to regulations that appear to contribute to the objectives of the NPR II study. While we have not reviewed the NEI proposal in detail, we believe the staff should give it appropriate consideration during the course of the NPR II study.

We wish to be informed of the results of the NPR II study.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

References:

1. Letter dated March 6, 1995, from NRC Chairman Ivan Selin, to Alice M. Rivlin, Director, Office of Management and Budget, regarding Nuclear Regulatory Commission's National Performance Review Phase II options paper
2. Memorandum dated March 7, 1995, from James M. Taylor, Executive Director for Operations, NRC, to K. Cyr, OGC, et al., Subject: National Performance Review Phase 2
3. Letter dated April 3, 1995, from William H. Rasin, Nuclear Energy Institute, to Jack Roe, Director, NRC NPR II Steering Committee, Subject: National Performance Review -- Phase 2
4. SECY-95-089 dated April 10, 1995, Memorandum from James M. Taylor, Executive Director for Operations, NRC, for the Commissioners, Subject: Semiannual Status Report on the Implementation of Regulatory Review Group Recommendations

5. U.S. Nuclear Regulatory Commission Administrative Letter 95-02 dated February 23, 1995, from Eugene V. Imbro, Office of Nuclear Reactor Regulation, Subject: Cost Beneficial Licensing Actions
6. ACRS report dated July 15, 1993, from J. Ernest Wilkins, Jr., Chairman, ACRS, to Ivan Selin, Chairman, NRC, Subject: Regulatory Review Group Report

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

February 17, 1995

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

Dear Mr. Taylor:

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

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

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

A number of utilities and NEI, which have commented on a draft version of this amendment, argue that it is overly prescriptive and contrary to the trend towards performance-based regulation. However, a suitable "metric," which could be used as the basis for a performance-based inspection for the assurance of the structural integrity of the containment, seems difficult to identify. Risk-based inspection appears to be a more promising approach to rationalizing in-service inspection of passive structural components. The Office of Nuclear Regulatory Research is actively pursuing this approach, and we hope to see risk-based concepts being used to develop requirements for in-service inspections in the not-too-distant future.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

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

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

March 14, 1995

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

Dear Chairman Selin:

SUBJECT: PROPOSED AMENDMENT TO THE NUCLEAR POWER PLANT LICENSE RENEWAL RULE (10 CFR PART 54)

During the 419th meeting of the Advisory Committee on Reactor Safeguards, March 9-10, 1995, we discussed with the NRC staff their proposal for the amendment to the License Renewal Rule (10 CFR Part 54). We also heard from representatives of the Nuclear Energy Institute and the Baltimore Gas and Electric Company on this matter. We had the benefit of the documents referenced.

The staff is proposing to revise the requirements contained in the rule to make it clearer and simpler and to allow more flexibility in its implementation. The intent of the amended rule continues to be to ensure that operation beyond the term of the original operating license will not jeopardize the public health and safety and that the current licensing basis will be preserved.

The amended rule is better integrated with the Maintenance Rule, and thereby provides greater coherence to the regulatory process. We agree that the proposed amendment to the current License Renewal Rule takes proper account of the existing licensee programs and provides a more stable and predictable license renewal process.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

References:

1. Memorandum dated February 9, 1995, from W. T. Russell, Chair, License Renewal Rule Steering Group, to J. T. Larkins, Executive Director, ACRS, Subject: Transmittal of the 10 CFR Part 54 Statements of Consideration and Rule Language Associated With the Amendment to the License Renewal Rule

2. Staff Requirements Memorandum dated June 24, 1994, from John C. Hoyle, Acting Secretary, to James M. Taylor, Executive Director for Operations, NRC, Subject: SECY-94-140 - Proposed Amendment to the Nuclear Power Plant License Renewal Rule (10 CFR Part 54)
3. SECY-94-140 dated May 23, 1994, from James M. Taylor, Executive Director for Operations, NRC, for the Commissioners, Subject: Proposed Amendment to the Nuclear Power Plant License Renewal Rule (10 CFR Part 54)

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

March 17, 1995

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

Dear Chairman Selin:

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

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

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

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

Sincerely,

T. S. Kress

T. S. Kress
Chairman

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

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

April 12, 1995

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

Dear Chairman Selin:

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

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

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

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

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

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

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

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

Sincerely,

T. S. Kress

T. S. Kress
Chairman

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

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

April 13, 1995

MEMORANDUM TO: James M. Taylor
Executive Director for Operations

FROM: John T. Larkins, Executive Director
Advisory Committee on Reactor Safeguards

SUBJECT: PROPOSED AMENDMENTS TO 10 CFR PART 73 CONCERNING SECURITY REQUIREMENTS ASSOCIATED WITH CONTAINMENT ACCESS CONTROL

During the 420th meeting of the Advisory Committee on Reactor Safeguards, April 6-7, 1995, the Committee decided not to comment on the proposed rule.

Reference:
Memorandum dated April 3, 1995, from Eric Beckjord, Director, RES, to William Russell, Director NRR, and others, Subject: Office Review and Concurrence on Proposed Rulemaking Changes to Nuclear Power Plant Security Requirements Associated with Containment Access Control (10 CFR Part 73)

cc: J. Hoyle, SECY
J. Blaha, OEDO
M. Taylor, OEDO
E. Beckjord, RES
B. Morris, RES
S. Bahadur, RES
A. Tse, RES
S. Frattali, RES
G. Sege, RES

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

April 13, 1995

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

Dear Chairman Selin:

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

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

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

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

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

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

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

Sincerely,

T. S. Kress

T. S. Kress
Chairman

<u>References</u>:

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

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

May 10, 1995

MEMORANDUM TO: James M. Taylor
Executive Director for Operations

FROM: John T. Larkins, Executive Director
Advisory Committee on Reactor Safeguards

SUBJECT: PROPOSED RULE CHANGES TO 10 CFR SECTION 50.55a, "CODES AND STANDARDS"

During the 421st meeting of the Advisory Committee on Reactor Safeguards, May 4-6, 1995, the Committee decided to review the proposed changes to 10 CFR 50.55a after the staff has reconciled the public comments. The Committee has no objection to the NRC staff proposal to issue the proposed rule for public comment.

Reference:
Memorandum dated April 17, 1995, from Frank Cherny, Office of Nuclear Reactor Regulation, to Brian Sheron, Office of Nuclear Reactor Regulation, and Lawrence Shao, Office of Nuclear Regulatory Research, Subject: Internal Review of Proposed Rule Changes, Title 10 of the Code of Federal Regulations, Section 50.55a, "Codes and Standards"

cc: J. Hoyle, SECY
J. Blaha, OEDO
M. Taylor, OEDO
B. Sheron, NRR
R. Wessman, NRR
J. Strosnider, NRR
G. Bagchi, NRR
F. Cherny, NRR
L. Shao, RES
M. Mayfield, RES
G. Sege, RES

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

June 16, 1995

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

Dear Mr. Taylor:

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

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

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

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

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

Sincerely,

T. S. Kress
Chairman

<u>References</u>:

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

   - Amendment to 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events"
   - Amendment to 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements"
   - Amendment to 10 CFR Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements"
   - Final Rule (10 CFR 50.66), "Requirements for Thermal Annealing of the Reactor Pressure Vessel"
   - Proposed Regulatory Guide 1.XXXX, "Format and Content of Report for Thermal Annealing of Reactor Pressure Vessels"

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

June 16, 1995

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

Dear Chairman Selin:

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

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

Sincerely,

T. S. Kress

T. S. Kress
Chairman

References:
1. Memorandum dated June 6, 1995, from Joseph A. Murphy, Executive Assistant to the Director, RES, to John T. Larkins, Executive Director, ACRS, Subject: ACRS Review of Final Amendment to Appendix J of 10 CFR Part 50 (Draft Predecisional Attachment)
2. ACRS Report dated September 19, 1994, from T. S. Kress, Chairman, ACRS, to The Honorable Ivan Selin, Chairman, NRC, Subject: Proposed Revisions to Appendix J of 10 CFR Part 50, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors"

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

September 15, 1995

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

Dear Mr. Taylor:

SUBJECT: THE NUCLEAR ENERGY INSTITUTE PETITION FOR RULEMAKING TO AMEND 10 CFR 50.48, "FIRE PROTECTION"

During the 424th meeting of the Advisory Committee on Reactor Safeguards, September 7-8, 1995, we completed our discussion regarding the subject rulemaking petition. Our Auxiliary and Secondary Systems Subcommittee met on June 7, 1995, to begin the review of this matter. During these meetings, we had the benefit of discussions with representatives of the staff, the Nuclear Energy Institute (NEI), and the Electric Power Research Institute (EPRI). We also had the benefit of the documents referenced.

The NEI petition for rulemaking proposes to amend 10 CFR 50.48, "Fire Protection," by adding an Appendix S, which is described as a "performance-based" alternative to the existing prescriptive Appendix R. NEI believes that the recommended addition to 10 CFR 50.48 will be "safety neutral" and that considerable cost savings will result.

We support risk-based regulations. It is not clear, however, how performance-based regulations should be developed from risk consideration. It is our perception that such regulations should include the following elements:

- Clearly stated objectives with demonstrable performance requirements, expressed either in deterministic or probabilistic terms.

- Flexibility in the methods that the licensee is permitted to use to meet the performance goals or criteria. These methods should be supported by operational experience and experimental results.

- The regulatory body must have a valid means to establish that the performance criteria have been met.

Unfortunately, the proposed rule in the NEI petition is deficient in all these elements.

The objective of the proposed rule is to assure "that the safety functions required to safely shut a plant down and maintain it in a safe condition are maintained during and following a fire." It is further stated that fire modeling, as well as PRAs, may be used to identify the pertinent performance criteria. The proposed rule, however, avoids setting probabilistic requirements and uses non-quantitative language. Thus, there are references to "credible" fires and "credible" scenarios, as well as to "adequate" time for completing safety functions. These concepts need to be defined in quantitative, probabilistic terms. For example, we would expect a quantitative performance requirement for the probability that fire will compromise safe shutdown equipment and lead to core damage.

Some of the issues that the proposed rule raises could be naturally resolved in a PRA context. Examples are the inadvertent actuation of automatic suppression systems and the relevance of the current requirements regarding the concurrent occurrence of a fire and loss of offsite power. In addition, the proposed rule does not address the issue of transient fuels. PRAs have shown that, in some cases, transient fuels are required to produce fires of severity sufficient to damage redundant safety systems. Such transient fuels have been found in controlled areas in the past. Not only are transient fuels not addressed, the proposed rule suggests that some administrative controls dictated by Appendix R may be eliminated. We would prefer to see an evaluation of such issues in the context of a fire PRA.

We are concerned that neither the NRC nor NEI has any plans for conducting fire tests for refining the probabilistic analysis of time-to-suppression. We also have concerns about weakening the requirement for automatic fire detection systems, the lack of a methodology for treating the potentially damaging effects of smoke, the use of a limited fire initiation database, and the neglect of consideration of fire during shutdown. We will address these concerns should the rulemaking process advance.

Even though we support the use of PRA in the development of a performance-based rule, we note that, given the uncertainties in the state of the art, fire PRAs cannot be the sole basis for regulatory requirements. Developing the right mix of criteria based on PRA and criteria based on good engineering practice is a challenge and a necessary requirement for a well-written rule.

We believe it will take some time and resources to develop and institute performance-based fire regulation. We also believe doing so is an important step in the agency's move in this direction.

Additional comments by ACRS Members George Apostolakis, James C. Carroll, and Ivan Catton are presented below.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

Additional Comments by ACRS Members George Apostolakis, James C. Carroll, and Ivan Catton

We support the Committee letter but have further comments for your consideration. The use of performance-based rules for fire protection is frustrated by conventional attitudes. The desire of regulators to have simple rules and tests for administrative convenience contrasts with the need of plant operators to have flexibility to arrive at optimal solutions. Unfortunately, the prescriptive characteristics embodied in regulations are accepted without proof, while any engineering solution supporting a performance requirement is subjected to a disproportionately higher standard of proof.

References:

1. Letter dated February 2, 1995, from W. Rasin, Nuclear Energy Institute, to John C. Hoyle, Acting Secretary, NRC, Subject: Petition for Rulemaking to Amend 10 CFR 50.48
2. SECY-94-090 dated March 31, 1994, from James M. Taylor, Executive Director for Operations, NRC, to the Commissioners, Subject: Institutionalization of Continuing Program for Regulatory Improvement
3. SECY-95-034 dated February 13, 1995, from James M. Taylor, Executive Director for Operations, NRC, to the Commissioners, Subject: Status of Recommendations Resulting from the Reassessment of the NRC Fire Protection Program
4. Memorandum dated December 30, 1994, from James M. Taylor, Executive Director for Operations, NRC, to the Commissioners, Subject: Eighth Quarterly Report on the Status of the Thermo-Lag Action Plan

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

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

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* For Items 1 through 14, see NUREG-1125, Volume 16, 4/95.

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) nas 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

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

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* For Items 1 through 14, see NUREG-1125, Volume 16, 4/95.

NRC FORM 335
(2-89)
NRCM 1102,
3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

1. REPORT NUMBER
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