NUREG-1125
Volume 18

A Compilation of
Reports of
The Advisory Committcee on Reactor Safeguards

1996 Annual

U.S. Nuclear Regulatory
Commission

April 1997

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ABSTRACT

This compilation contains 47 ACRS reports submitted to the Commission, or to the Executive Director for Operations, during calendar year 1996. 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, the U. S. Library of Congress, and the Internet at http://www.nrc.gov/ACRSACNW. 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 1996. 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
17	Calendar Year 1995

ACRS MEMBERSHIP (1996)

CHAIRMAN:	Dr. Thomas S. Kress, Retired Oak Ridge National Laboratory
VICE CHAIRMAN:	Dr. Robert L. Seale, Professor Emeritus University of Arizona
MEMBERS:	Dr. George Apostolakis, Professor Massachusetts Institute of Technology
	Mr. John J. Barton, Retired GPU Nuclear Corporation
	Mr. James C. Carroll, Retired Pacific Gas & Electric Company **(Term ended 5/96)**
	Dr. Ivan Catton, Professor University of California, Los Angeles
	Dr. Mario H. Fontana, Research Professor University of Tennessee
	Mr. William J. Lindblad, Retired Portland General Electric Company **(Term ended 9/96)**
	Dr. Don W. Miller, Professor The Ohio State University
	Dr. Dana A. Powers Sandia National Laboratories
	Dr. William J. Shack Argonne National Laboratory
	Mr. Charles J. Wylie, Retired Duke Power Company **(Term ended 9/96)**

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

November 18, 1996

MEMORANDUM TO: James M. Taylor
Executive Director for Operations

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

SUBJECT: DRAFT REPORTS RELATED TO THE KEOWEE HYDRO STATION EMERGENCY ELECTRICAL SYSTEM SUPPLY TO THE OCONEE NUCLEAR STATION

During the 436th meeting of the Advisory Committee on Reactor Safeguards, November 7-9, 1996, the Committee considered the subject reports and the licensee's schedule for implementing the proposed modifications described in these reports. The Committee decided not to review these reports at this time. The Committee, however, may hear a briefing after the licensee has completed the proposed modifications.

Reference:
Letter dated July 8, 1996, from W. T. Russell, Director, NRR, to J.W. Hampton, Vice President, Oconee Site, Duke Power Company, Subject: Draft Reports Related to the Keowee Hydro Station Emergency Electrical System Supply to the Oconee Nuclear Station

cc: J. Hoyle, SECY
J. Blaha, OEDO
J. Mitchell, OEDO
E. Jordan, AEOD
A. Thadani, NRR
F. Hebdon, NRR
J. Cortez, RES

Part 2: ACRS Reports on Generic Subjects

2

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

March 19, 1996

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

Dear Mr. Taylor

SUBJECT: NRC STAFF PROGRAM ON THE ADEQUACY ASSESSMENT OF THE RELAP5/MOD3 CODE FOR SIMULATION OF AP600 PASSIVE PLANT BEHAVIOR

During the 429th meeting of the Advisory Committee on Reactor Safeguards, March 7-9, 1996, we reviewed the program being conducted by the Office of Nuclear Regulatory Research (RES) to assess the adequacy of the RELAP5/MOD3 code for simulating the behavior of the Westinghouse AP600 passive plant design. During this review, we had the benefit of discussions with representatives and consultants of the NRC staff and the Idaho National Engineering Laboratory (INEL). Our Subcommittee on Thermal Hydraulic Phenomena held a meeting on this matter on February 22-23, 1996. We also had the benefit of the referenced documents.

We have been asked to comment on the approach and methodology for demonstrating the adequacy of the RELAP5/MOD3 code to calculate AP600 passive plant behavior in support of the design certification review. We believe that the overall approach and methodology being employed by RES for this assessment is acceptable. Most of the necessary elements are in place. A substantial amount of work remains, however, and we believe that the schedule for successful completion cannot be met.

Our comments and recommendations relative to this review, primarily based on oral presentations, are:

- Since we last reviewed this program in 1994, significant improvements have been made. The most significant has been the increased emphasis on the code improvement program. Other changes that have led to excellent results include the involvement of outside technical expertise, via the Thermal Hydraulic Expert Consultants group and the direct involvement of RES technical personnel in the research activities. Particularly noteworthy accomplishments include the analysis of water hammer, the treatment of flow oscillations observed in the tests during injection from the In-containment Refueling-Water Storage Tank and the evaluation and explanation of strong thermal stratification in the ROSA cold leg.

- RES should perform a more robust and complete top-down system scaling analysis for ROSA, SPES, and OSU. An entire transient should be evaluated to quantify the effects of various distortions in the three facilities and to demonstrate that the experimental database is sufficient to validate the code. Any additional distortions or anomalies identified should be added to the list of distortions compiled by RES in late-1994, and that remain to be addressed. The scaling effort should be integrated with the Phenomena Identification and Ranking Table.

- The thermal stratification that was seen in ROSA tests for a one-inch cold-leg break was initially identified as a potentially important safety issue for the AP600. It has now been shown to be just a manifestation of scale distortion in the ROSA facility. This demonstrates the need to identify and explain anomalous behavior.

- The thermal stratification in the Core Makeup Tank (CMT) observed in the tests needs to be studied. Its effects on core inventory have to be understood because neither RELAP5/MOD3 nor the Westinghouse computer codes can, at present, reliably predict thermal stratification.

- The screening study for water hammer in the AP600 design addressed an important safety issue. The study allows an analysis of the potential for such events and provides a method for estimating the resulting loads in susceptible areas. We recommend that this study be published soon as a separate report.

- The documentation provided for our review did not, by itself, furnish an adequate basis upon which we could logically endorse the process. The documentation provided to the Thermal Hydraulic Phenomena Subcommittee in advance of the February 22-23, 1996 meeting was inconsistent and contained results declared incorrect by RES during the meeting. Furthermore, the RELAP5/MOD3 Code Manual published in August 1995 was not provided to us in time to support our review.

- RELAP5 is still undergoing significant and rapid modifications. A calculation has not yet been performed with a version of the code that contains all the planned changes. Numerous calculations will need to be performed to mature the code and validate it using data obtained from various separate effects and integral facilities tests.

Overall, the approach and methodology for qualifying RELAP5/MOD3 for AP600 simulation appear to be adequate. However, two possible "show stoppers" remain: 1) simulation of the CMT thermal stratification and 2) simulation of long-term cooling, which is still an issue. Serious consideration should be given to addressing these obstacles.

Dr. George Apostolakis did not participate in the Committee's deliberations of this matter.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

References:

1. Memorandum dated January 22, 1996 from M. W. Hodges, Office of Nuclear Regulatory Research, NRC, to J. Larkins, Advisory Committee on Reactor Safeguards, NRC, transmitting:

   - Volume 2 of 10 volumes of adequacy demonstration reports, "Adequacy Assessment Overview"

   - Idaho National Engineering Laboratory draft report prepared for U.S. Nuclear Regulatory Commission, "Adequacy Evaluation of RELAP5/MOD3 for Simulating AP600 Small Break Loss-of-Coolant Accidents, Volume 2: Horizontal Integrated Analysis of the AP600 1-Inch Diameter Cold Leg Break," November 1995, with Appendices A-K (Proprietary)

2. Idaho National Engineering Laboratory, draft report prepared for U.S. Nuclear Regulatory Commission, "Top-Down Scaling Analysis Methodology for AP600 Integral Tests," January 1996

3. Letter report dated April 12, 1995, to James M. Taylor, Executive Director for Operations, NRC, from T. S. Kress, Chairman, Advisory Committee on Reactor Safeguards, Subject: NRC Test and Analysis Program in Support of AP600 Advanced Light Water Passive Plant Design Review

4. Letter dated May 8, 1995, from James M. Taylor, Executive Director for Operations, NRC, to T. S. Kress, Chairman, Advisory Committee on Reactor Safeguards, Subject: Staff Response to ACRS Letter Dated April 12, 1995, on NRC Test and Analysis Program in Support of AP600 Advanced Light Water Passive Plant Design Reviews

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

August 14, 1996

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

Dear Chairman Jackson:

SUBJECT: DESIGN CHANGES PROPOSED BY ASEA BROWN BOVERI - COMBUSTION ENGINEERING RELATING TO THE CERTIFICATION OF THE SYSTEM 80+ DESIGN

During the 433rd meeting of the Advisory Committee on Reactor Safeguards, August 8-10, 1996, we reviewed recent design changes proposed by ASEA Brown Boveri - Combustion Engineering (ABB-CE) relating to the certification of the System 80+ design. These "design changes" consist of both actual modifications to the design and corrections to the documentation to remove inconsistencies and typographical errors. We had the benefit of discussions with representatives of the NRC staff and of ABB-CE. We also had the benefit of the documents referenced.

Conclusions

Our review of Supplement 1 to NUREG-1462, "Final Safety Evaluation Report Related to the Certification of the System 80+ Design," did not change the conclusion reached in our earlier report of May 11, 1994. We continue to believe that acceptable bases and requirements have been established in the application to assure that the System 80+ Standard Design can be used to engineer and construct plants that with reasonable assurance can be operated without undue risk to the health and safety of the public.

Background and Discussion

We have been involved in the review of the System 80+ design since ABB-CE applied for certification. This review was carried out in accordance with 10 CFR Part 52, which requires ACRS to report on those portions of 10 CFR Part 52 applications that concern safety. In our May 11, 1994 report to the Commission, we supported the certification of the System 80+ design. This report was included in the staff Safety Evaluation Report (NUREG-1462). The present review is intended to supplement our earlier review of this ABB-CE application.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

References:
1. U. S. Nuclear Regulatory Commission, NUREG-1462, Supplement No. 1, "Final Safety Evaluation Report Related to the Certification of the System 80+ Design," dated July 1, 1996
2. ACRS Report dated May 11, 1994, from T. S. Kress, Chairman, ACRS, to Ivan Selin, Chairman, NRC, Subject: Report on the Safety Aspects of the ASEA Brown Boveri-Combustion Engineering Application for Certification of the System 80+ Standard Plant Design
3. Letter dated June 27, 1996, from C. B. Brinkman, ABB-Combustion Engineering Nuclear Systems, to U.S. Nuclear Regulatory Commission, regarding System 80+ Standard Plant Design Changes
4. Letter dated July 17, 1996, from C. B. Brinkman, ABB-Combustion Engineering Nuclear Systems, to U.S. Nuclear Regulatory Commission, regarding six additional design changes for System 80+ Standard Plant Design

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

August 15, 1996

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

Dear Chairman Jackson:

SUBJECT: SECY-96-128, "POLICY AND KEY TECHNICAL ISSUES PERTAINING TO THE WESTINGHOUSE AP600 STANDARDIZED PASSIVE REACTOR DESIGN"

During the 433rd meeting of the Advisory Committee on Reactor Safeguards, August 8-10, 1996, we reviewed the subject document. Our Subcommittee on Westinghouse Standard Plant Designs met on July 19, 1996 to review this matter. During this review, we had the benefit of discussions with representatives of the staff and of the Westinghouse Electric Corporation. We also had the benefit of the documents referenced.

Conclusion

We endorse the positions recommended by the staff in addressing the following three policy issues pertaining to the Westinghouse AP600 standardized passive reactor design.

Policy Issues

- Prevention and Mitigation of Severe Accidents

  The staff is seeking Commission approval to consider the use of non-safety systems in the AP600 design to address the uncertainties associated with the passive fission product removal mechanisms for design-basis analysis and for balance between prevention and mitigation of severe accidents. Westinghouse has no objection to the staff's crediting of non-safety equipment that is already a part of the AP600 design, but objects to a requirement for adding a non-safety-grade containment spray system.

  The applicant's submittals provide some support for demonstrating fission product removal using only passive removal mechanisms. Nonetheless, we are persuaded by the staff position that systems beyond the passive removal mechanisms should be evaluated to provide greater confidence in the performance of the plant design in mitigating design-basis and severe accidents. We recommend Commission approval.

- External Reactor Vessel Cooling

  The staff is seeking Commission approval for requiring that the applicant provide limited analytical evaluation of postulated ex-vessel phenomena, notwithstanding that the AP600 design is intended to prevent reactor vessel melt-through. We recommend Commission approval.

- Post-72-hour Actions

  The staff is seeking Commission approval for requiring that the AP600 design be capable of sustaining all design-basis events with onsite equipment and supplies for the long term. We recommend Commission approval.

Technical Issues

The staff added spent fuel pool cooling to its list of technical issues being tracked in the review. At present, the applicant will be required to provide additional onsite capability to remove decay heat from the spent fuel pool over an extended period of time. We believe this requirement may be found unnecessary after considering the low risk associated with the current design.

Dr. Dana A. Powers did not participate in the Committee's deliberations regarding the severe accident source term. Dr. T. 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. U. S. Nuclear Regulatory Commission, SECY-96-128, dated June 12, 1996, from James M. Taylor, Executive Director for Operations, NRC, to the Commissioners, Subject: Policy and Key Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design
2. Letter dated June 15, 1995, from T.S. Kress, Chairman, ACRS, to James M. Taylor, Executive Director for Operations, NRC, Subject: Proposed Commission Paper on Staff Positions on Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design
3. Letter dated August 8, 1995, from James M. Taylor, Executive Director for Operations, NRC, to T.S. Kress, Chairman, ACRS, Subject: Response to ACRS Comments on Commission Paper on Technical Issues Pertaining to the Westinghouse AP600 Design

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

August 15, 1996

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

Dear Chairman Jackson:

SUBJECT: DESIGN CHANGES PROPOSED BY GENERAL ELECTRIC NUCLEAR ENERGY RELATING TO THE CERTIFICATION OF THE U.S. ADVANCED BOILING WATER REACTOR DESIGN

During the 433rd meeting of the Advisory Committee on Reactor Safeguards, August 8-10, 1996, we reviewed recent design changes proposed by General Electric Nuclear Energy (GENE) relating to the certification of the U.S. advanced boiling-water reactor (ABWR) design. These "design changes" consist of both actual modifications to the design and corrections to the documentation to remove inconsistencies and typographical errors. We had the benefit of discussions with representatives of the NRC staff and of GENE. We also had the benefit of the documents referenced.

Conclusions

Our review of Supplement 1 to NUREG-1503, "Final Safety Evaluation Report Related to the Certification of the U.S. ABWR Design," did not change the conclusion reached in our earlier report of April 14, 1994. We continue to believe that acceptable bases and requirements have been established in the application to assure that the U.S. ABWR Standard Design can be used to engineer and construct plants that with reasonable assurance can be operated without undue risk to the health and safety of the public.

Background and Discussion

We have been involved in the review of the U.S. ABWR design since GENE applied for certification. This review was carried out in accordance with 10 CFR Part 52, which requires ACRS to report on those portions of 10 CFR Part 52 applications that concern safety. In our April 14, 1994 report to the Commission, we supported the certification of the U.S. ABWR design. This report was included in the staff Safety Evaluation Report (NUREG-1503). The present review is intended to supplement our earlier review of this ABWR application.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

References:
1. U. S. Nuclear Regulatory Commission, NUREG-1503, Supplement No. 1, "Final Safety Evaluation Report Related to the Certification of the Advanced Boiling Water Reactor Design," dated July 1, 1996
2. Staff Requirements Memorandum dated June 11, 1996, from John C. Hoyle, Secretary, to John T. Larkins, ACRS, regarding meeting with Advisory Committee on Reactor Safeguards, May 24, 1996
3. ACRS Report dated April 14, 1994, from T. S. Kress, Chairman, ACRS, to Ivan Selin, Chairman, NRC, Subject: Report on the Safety Aspects of the General Electric Nuclear Energy Application for Certification of the Advanced Boiling Water Reactor Design
4. Letter dated April 16, 1996, from J. F. Quirk, GE Nuclear Energy, to Dennis M. Crutchfield, Nuclear Regulatory Commission, regarding ABWR design changes
5. Letter dated July 1, 1996, from J. F. Quirk, GE Nuclear Energy, to the Nuclear Regulatory Commission, Subject: ABWR Design Control Document Changes

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

February 23, 1996

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

Dear Chairman Jackson:

SUBJECT: WESTINGHOUSE BEST-ESTIMATE LOSS-OF-COOLANT ACCIDENT ANALYSIS METHODOLOGY

During the 428th meeting of the Advisory Committee on Reactor Safeguards, February 8-10, 1996, we reviewed the best-estimate, large-break, loss-of-coolant accident (LBLOCA) analysis methodology developed by the Westinghouse Electric Corporation. During this review, we had the benefit of discussions with representatives of the NRC staff, Westinghouse, Idaho National Engineering Laboratory, and several nuclear power plant licensees. Our Subcommittee on Thermal Hydraulic Phenomena has held a number of meetings on this matter as far back as 1991. The last meeting of the Subcommittee concerning this issue was held on January 18-19, 1996. We also had the benefit of the referenced documents.

Westinghouse has developed an improved method to evaluate the performance of emergency core cooling systems (ECCS) for the case of a LBLOCA in three- and four-loop pressurized-water reactors (PWRs) of Westinghouse design. Westinghouse has proposed that this improved method, based on the use of the WCOBRA/TRAC code, be accepted for routine use in demonstrating that the cores in these plants meet NRC licensing requirements pursuant to the revised ECCS Rule (10 CFR 50.46). The NRC staff has reviewed this proposal and has concluded that the new methodology can be used for licensing calculations. We concur with the staff; however, some improvements in the uncertainty analysis are desirable.

The improved method of analysis takes advantage of data and the understanding of thermal-hydraulic behavior developed during the past two decades. This method will reduce the conservative margins in the calculated peak cladding temperature that result from the use of current methods based on Appendix K. This will permit licensees of Westinghouse three- and four-loop PWRs to have greater flexibility in the operation of their plant reactor cores and in associated fuel management practices. We also believe that, when properly documented, the improved method will provide a straight-

forward and understandable assessment of the performance of an important safety system.

The improved LOCA evaluation method makes use of realistic values for inputs and correlations rather than the conservatively biased values used in the past. To meet licensing requirements, empirically based uncertainty distributions for each of the important inputs and correlations are used and propagated through the solution algorithm, WCOBRA/TRAC, to obtain estimates of uncertainty distributions for the peak cladding temperature. A nominal 95 percent probability of nonexceedance is required for licensing purposes. Questionable models or correlations are adjusted to ensure that their predictions are conservative. Westinghouse expanded the Code Scaling, Applicability, and Uncertainty (CSAU) evaluation methodology outlined in NRC Regulatory Guide 1.157, by including additional parameters not considered during the earlier CSAU exercise conducted by the NRC staff.

We have some concerns about the Westinghouse best-estimate LBLOCA evaluation methodology. The method used by Westinghouse to obtain the heat transfer coefficient uncertainty distribution resulted in some high values that are nonphysical. Westinghouse should reevaluate the heat transfer uncertainty distribution with appropriate consideration of the dependencies on physical parameters such as reflood rate. The Westinghouse treatment of the minimum wetting (or rewetting) temperature is not satisfactory because the correlation ignores important phenomena and could lead to nonconservative results. The existence of compensating errors in WCOBRA/TRAC may be a reason for the skewed distribution in heat transfer coefficients. We believe that these concerns should be addressed.

Obtaining adequate documentation in a timely manner has been a problem from the outset of this review. This has unnecessarily complicated the reviews by both the NRC staff and the ACRS. Westinghouse has committed to provide documentation that will clearly lay out its LBLOCA methodology. We believe that the staff should review this final documentation prior to approving use of the improved methodology. The staff should also prepare guidelines for documentation of future best-estimate LOCA submittals before the lessons learned from this review are forgotten.

It is important to realize that the deficiencies seen in codes like TRAC and RELAP may preclude their extension to the evaluation of best-estimate ECCS performance under small-break LOCA conditions or to passive plant designs. The use of WCOBRA/TRAC is acceptable for LBLOCA calculations because of the extensive test data available for code validation and the associated analytical expertise developed over the past 20 years. A comparable database does not exist for many other applications.

We commend the staff and Westinghouse for completing an important task. If the above concerns are adequately addressed, the result will be a much improved best-estimate method for the prediction of LBLOCA behavior in light-water reactors.

ACRS Member George Apostolakis did not participate in the Committee's deliberation of this matter.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

References:

1. Westinghouse Topical Report, "Code Qualification Document for Best Estimate LOCA Analysis," WCAP-12945-P, Revision 1, Volumes 1-5, June 1992 (Proprietary)
2. U. S. Nuclear Regulatory Commission, "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Acceptability of the Topical Report WCAP-12945(P) 'Westinghouse Code Qualification Document for Best-Estimate Loss of Coolant Accident Analysis' for Referencing in PWR Licensing Applications, Westinghouse Electric Corporation" (Draft) and "Draft Technical Evaluation Report, Westinghouse Code Qualification Document for Best Estimate Loss of Coolant Accident Analyses, WCAP-12945-P" (Proprietary), transmitted by P. Boehnert, ACRS staff, to the ACRS Thermal Hydraulic Phenomena Subcommittee, by memorandum dated January 4, 1996
3. Memorandum, dated November 3, 1995, from P. Boehnert, ACRS staff, to I. Catton, Chairman, ACRS Thermal Hydraulic Phenomena Subcommittee, Subject: NRC/NRR-Westinghouse Meeting, October 23-24, 1995 - Westinghouse Best-Estimate ECCS Evaluation Model Code, WCOBRA/TRAC", including W memorandum, dated October 13, 1995, transmitting "Revisions to the W Best-Estimate Uncertainty Methodology" (Proprietary)
4. Memorandum dated January 5, 1996 from M. Nissley, Westinghouse, to Members and Consultants of the ACRS Thermal Hydraulic Phenomena Subcommittee, transmitting the following reports:
   - NTD-NRC-95-4505 - Roadmap Comparison with CSAU Methodology
   - NTD-NRC-95-4575 - Revised Uncertainty Methodology Report (Proprietary)
   - NTD-NRC-95-4586 - Assessment of Compensating Errors (Proprietary)
   - NTD-NRC-95-4588 - Non-Proprietary Executive Summary

4
- NTD-NRC-96-4618 - Responses to Several Issues Identified in INEL's Review of NTD-NRC-95-4575 (Proprietary)

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

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

February 26, 1996

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

Dear Chairman Jackson:

SUBJECT: PROPOSED FINAL NRC BULLETIN 96-XX, "POTENTIAL PLUGGING OF EMERGENCY CORE COOLING SUCTION STRAINERS BY DEBRIS IN BOILING WATER REACTORS" AND AN ASSOCIATED DRAFT REVISION 2 OF REGULATORY GUIDE 1.82, "WATER SOURCES FOR LONG-TERM RECIRCULATION COOLING FOLLOWING A LOSS-OF-COOLANT ACCIDENT"

During the 428th meeting of the Advisory Committee on Reactor Safeguards, February 8-10, 1996, we heard presentations by and held discussions with representatives of the NRC staff and the Boiling Water Reactor Owners Group (BWROG) concerning the proposed final Bulletin and the Revision 2 to Regulatory Guide 1.82. We also had the benefit of the documents referenced.

The emergency core cooling system (ECCS) strainer blockage event was initially raised following an event at Barsebäck Unit 2 in Sweden on July 28, 1992. The event involved containment spray system strainer blockage caused by debris dislodged as a result of a safety valve discharge and the activation of the drywell sprays. Subsequently, three strainer blockage events occurred at U.S. nuclear power plants: two at the Perry Nuclear Power Plant in April and November 1993, and one at the Limerick Plant in September 1995. If strainer blockage is coupled with a sustained loss-of-coolant accident (LOCA), the potential exists for serious core damage due to the impairment of plant emergency core cooling systems.

We were briefed previously by the staff on its response to the Barsebäck event in January 1993, July 1993, April 1994, and October 1994. In our report dated October 14, 1994, we expressed a concern about the slow pace of NRC and industry actions in response to this important safety issue. The staff planned to provide prescriptive design information for BWR suppression pool strainers in a revision to Regulatory Guide 1.82 similar to that provided in the current version of this Regulatory Guide for pressurized-water reactor (PWR) ECCS sumps (design sketches, dimensions, etc.). We questioned this approach and stated that the onus should be on the

BWR licensees to evaluate the vulnerability of their plants to ECCS strainer blockage due to LOCA-generated debris and to propose appropriate modifications to deal with this plant-specific issue. The staff reviewed our concern and concluded that its action plan for resolving this issue was appropriate. The Executive Director for Operations did, however, ask the staff to accelerate its resolution schedule to the extent practicable.

The staff believes continued operation of BWRs is acceptable while the actions requested in proposed Bulletin 96-XX are being implemented. This belief is based on the assessment that licensees have adequately responded to Bulletin 93-02 and its supplement and to Bulletin 95-02, which required interim actions to minimize foreign materials from drywells and suppression chambers that could clog ECCS strainers.

Proposed Bulletin 96-XX requires all BWR licensees (except for Big Rock Point, which has a dry containment) to submit a report, within 180 days of issuance of the Bulletin, detailing their planned actions. Licensees would then be required to complete needed plant modifications before the end of the first refueling outage following their submittal.

The staff has identified three resolution options:

- Installation of large capacity passive strainers
- Installation of self-cleaning strainers
- Installation of strainer backflush systems and associated instrumentation alarms and operator training in the use of the system

Both the staff and BWROG prefer the first option, but realize that it may be difficult for some licensees to provide the structural support needed for LOCA-induced hydrodynamic loads.

The staff will allow licensees to propose other solutions. (A licensee may also propose no action, but must provide a detailed description of the safety basis for its decision.) Licensees must propose suitable Technical Specifications for the surveillance requirements for their planned actions. Both the staff and the BWROG agree that the potential for ECCS strainer blockage following a LOCA is a compliance issue. Accordingly, the staff will require the use of safety-grade equipment in any plant modifications that are made unless a licensee can provide a suitable technical basis for using nonsafety-grade equipment.

We have a number of observations regarding the present status of the resolution of this issue:

- Each of the options described above requires that strainer debris loading be calculated in accordance with the proposed Revision 2 to Regulatory Guide 1.82. This Regulatory Guide, however, only delineates the phenomena that should be considered in calculating strainer debris loading. The staff has told the BWROG that an additional year will be necessary for the staff to develop the calculational methodology to evaluate the performance of existing and retrofit strainer designs. The staff has stated that the purpose of this effort is to be able to respond to anticipated licensee responses to the Bulletin. This is a major change from the earlier staff position that it would provide prescriptive information for the design of BWR ECCS strainers in the revision to the Regulatory Guide.

- The BWROG has performed extensive analytical and experimental work and has developed and tested several potential hardware modifications, including improved passive strainer designs and a self-cleaning strainer. Documentation will be completed and submitted to the NRC over the next few months. The BWROG is also developing a guidance document to assist licensees in complying with the final Bulletin and Regulatory Guide. This document is scheduled for completion in June 1996. The staff is committed to promptly review and comment on this document.

- It may not be possible to predict with confidence the character and amount of debris that would challenge ECCS strainers. Strainers would still be susceptible to common-mode failure. A diverse means of providing emergency core cooling is desirable. The revised Regulatory Guide provides guidance for the licensees to review, and improve where required, the procedures related to core cooling from alternative sources of water. We believe that this is an important aspect of the resolution to the problem.

We agree with the staff that the Bulletin and revised Regulatory Guide should be issued as soon as possible in order to move toward resolution of this issue. The BWROG has not had an opportunity to review these documents in detail, but appears to be in general agreement with this course of action. Continued, close interaction between the staff and the BWROG will be needed to bring this issue to timely closure.

We note that the staff is reviewing the need for further action for PWRs beyond that taken in the 1985 resolution of Unresolved Safety Issue A-43, "Containment Emergency Sump Performance." We believe that this is appropriate in light of what has been learned about debris generation and transport.

Finally, we continue to believe that the response of the staff and the BWR licensees to this important nuclear safety issue has been unacceptably slow. We have asked the staff to keep us informed of the activities to bring this matter to closure.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

References:
1. Memorandum dated January 23, 1996, from F. Miraglia, Office of Nuclear Reactor Regulation, NRC, to E. Jordan, Committee to Review Generic Requirements, NRC, Subject: Request for Review and Endorsement of the Proposed Bulletin Titled, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling Water Reactors" (Draft Predecisional)
2. Memorandum dated January 5, 1996, from L. Shao, Office of Nuclear Regulatory Research, NRC, to J. Larkins, ACRS, Subject: ACRS Review of Regulatory Guide 1.82, Revision 2, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident"
3. U.S. Nuclear Regulatory Commission, NRC Bulletin 95-02, "Unexpected Clogging of a Residual Heat Removal (RHR) Pump Strainer While Operating in Suppression Pool Cooling Mode," dated October 17, 1995
4. U.S. Nuclear Regulatory Commission, NRC Bulletin 93-02, "Debris Plugging of Emergency Core Cooling Suction Strainers," dated May 11, 1993
5. Letter dated October 14, 1994, from T. S. Kress, Chairman, ACRS, to Ivan Selin, Chairman, NRC, Subject: Potential for BWR ECCS Strainer Blockage due to LOCA Generated Debris
6. Letter dated January 27, 1995, from James Taylor, Executive Director for Operations, NRC, to T. S. Kress, Chairman, ACRS, Subject: Potential for BWR ECCS Strainer Blockage due to LOCA Generated Debris

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

April 19, 1996

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

Dear Chairman Jackson:

SUBJECT: WESTINGHOUSE BEST-ESTIMATE LOSS-OF-COOLANT ACCIDENT ANALYSIS METHODOLOGY

During the 430th meeting of the Advisory Committee on Reactor Safeguards, April 11-13, 1996, we concluded our review of the best-estimate, large-break, loss-of-coolant accident (LBLOCA) analysis methodology developed by the Westinghouse Electric Corporation. We had previously reviewed this matter during our 428th meeting, February 8-10, 1996. We also had the benefit of the referenced documents.

In our February 23, 1996 report commenting on the results of our initial review, we identified several technical details of the Westinghouse LBLOCA methodology needing further attention and also commented on the adequacy of the documentation. As a result of subsequent discussions with representatives of Westinghouse and the NRC staff during this meeting, we believe that these concerns have been addressed.

ACRS Member George Apostolakis did not participate in the Committee's deliberation of this matter.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

<u>References</u>:

1. Memorandum dated March 25, 1996, from N. Liparulo, Westinghouse, to Nuclear Regulatory Commission, transmitting information on the resolution of issues related to the review of WCAP-12945-P (Proprietary)
2. Letter dated March 15, 1996, from J. Taylor, Executive Director for Operations, NRC, to T. S. Kress, Chairman, ACRS,

Subject: Westinghouse Best-Estimate Loss-of-Coolant Accident Analysis Methodology

3. Report dated February 23, 1996, from T. S. Kress, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: Westinghouse Best-Estimate Loss-of-Coolant Accident Analysis Methodology

March 15, 1996

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

Dear Chairman Jackson:

SUBJECT: REVIEW OF RECENT FIRE PROBABILISTIC RISK ASSESSMENT REPORTS BY BROOKHAVEN NATIONAL LABORATORY AND CERTAIN FIRE BARRIER ISSUES

During the 429th meeting of the Advisory Committee on Reactor Safeguards, March 7-9, 1996, we reviewed scoping fire probabilistic risk assessments (PRAs) performed by Brookhaven National Laboratory (BNL). We had the benefit of discussions with representatives of the staff, BNL, and the National Institute of Standards and Technology (NIST). Our Subcommittee on Fire Protection discussed this matter during a meeting on February 29, 1996. We also had the benefit of the documents referenced.

At your request, we reviewed both the PRA model that evaluated the strategy of using self-induced station blackout (SISBO) to mitigate the consequences of a fire in the control room or cable spreading room and the PRA-based scoping analysis of degraded fire barriers. We also discussed the development of alternate time-temperature curves for qualification of fire barriers and the status of other fire protection issues.

To comply with Appendix R requirements, eight units have procedures that require initiating a station blackout (SBO) condition. An additional fifteen units have procedures for dealing with fires in critical areas that could result in an SBO. The PRA by BNL evaluated the effects of different schemes for managing the electrical systems in the plant when a fire in the control room has required use of the alternate shutdown panel.

The study focused on the effectiveness of the procedures used to mitigate the fire and did not address the probabilistic treatment of fires. The scope of the study did not include a number of issues that could affect the conclusions. For example, the BNL study addressed neither the effects of fire and smoke on human actions nor the possible damage to sensitive electronic control and safety instrumentation. The study is weak in the areas of modeling human actions for the manual shutdown and restart of electrical equipment after an SBO condition. Because of the limitations of

the analysis and the failure to quantify uncertainties, no substantive conclusions can be drawn from this scoping study. The limitations of the analysis should be addressed in Phase 2 of this study. A meaningful uncertainty analysis should also be performed.

In the analysis of degraded fire barriers, BNL developed core-damage frequencies for fire scenarios involving failures of fire protection features such as cable tray fire barriers, automatic detection and suppression systems, and fire barrier penetrations. The PRA model did not examine degrees of fire barrier degradation.

The analysis was based on event tree/fault tree models. Although this is a step in the right direction, the analysis does not use the best available methods for modeling fire propagation, detection, and suppression. It does not model the fundamental competition between the time to damage and the time to detection/suppression. Most current fire PRAs have adopted the competing processes model.

We also discussed the program proposed to the staff by NIST to develop alternate time-temperature curves for nuclear power plant fire barrier qualification. The program includes development of models, ASTM E119-type full-scale furnace tests, and test methods to simulate barrier response. We question the need for this program. We have been told that alternate time-temperature curves have been produced by the insurance industry. Furthermore, a large number of fire models exist, some of which are being evaluated by the Department of Energy. Although the need for new models is not clear, more validation of these models with experimental data is needed. Some data exist (NUREG/CR-6017). Comparisons with fire model simulations show that the results are very sensitive to input parameters that are not always well known.

The staff summarized the progress of licensee actions to correct deficiencies associated with Thermo-Lag fire barriers. The program appears to be meeting its objectives.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

References:
1. Brookhaven National Laboratory, Draft Technical Letter Report, FIN L-2629, "Risk Evaluation of the Response of PWRs to Severe Fires in Critical Locations," May 30, 1995 (Draft Predecisional)

2. Brookhaven National Laboratory, Technical Evaluation Report, FIN L-1311, "A Risk-Based Approach for Evaluation of Fire Mitigation Features in Nuclear Power Plants," November 21, 1995 (Draft Predecisional)
3. U. S. Nuclear Regulatory Commission, NUREG/CR-6017 and SAND93-0528, "Fire Modeling of the Heiss Dampf Reaktor Containment," September 1995

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

February 22, 1996

MEMORANDUM TO: James M. Taylor
Executive Director for Operations

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

John T. Larkins

SUBJECT: PROPOSED GENERIC LETTER 96-XX ON PERIODIC VERIFICATION OF DESIGN-BASIS CAPABILITY OF SAFETY-RELATED MOTOR-OPERATED VALVES

During the 428th meeting of the Advisory Committee on Reactor Safeguards, February 8-10, 1996, the Committee decided not to review the subject proposed Generic Letter 96-XX. The Committee appreciates being afforded the opportunity to review the subject matter.

Reference:
Memorandum dated January 22, 1996, from Frank J. Miraglia, Jr., NRR, to Edward L. Jordan, Chairman, Committee to Review Generic Requirements, Subject: Proposed Generic Letter 96-XX on Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves

cc: J. Hoyle, SECY
J. Blaha, OEDO
L. Soffer, OEDO
B. Sheron, NRR
R. Wessman, NRR
J. Cortez, RES

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

March 14, 1996

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 78, "MONITORING OF FATIGUE TRANSIENT LIMITS FOR THE REACTOR COOLANT SYSTEM"

During the 429th meeting of the Advisory Committee on Reactor Safeguards, March 7-9, 1996, we completed our deliberations on the resolution of the subject Generic Safety Issue that we started during our 424th meeting, September 7-8, 1995. We had the benefit of discussions with representatives of the NRC staff and of the documents referenced.

This Generic Safety Issue was originally developed to determine whether licensees need to perform transient monitoring to ensure compliance with requirements concerning fatigue failure. The transient monitoring concern was subsumed in the Fatigue Action Plan, which was reported as complete in SECY-95-245, "Completion of the Fatigue Action Plan."

The current scope of the Generic Safety Issue is focused on the evaluation of risk from fatigue failure. The staff completed a study that demonstrated that the risk from fatigue failure of the primary coolant pressure boundary components is very small. The analyses used in the study were based on the assumption that the probability of crack initiation by fatigue in a component subject to cyclic loads and the probability of crack propagation through the wall are independent. The product of these probabilities was used to calculate the change in core-damage frequency caused by fatigue failure of a component.

The analyses, as presented to us by the staff to demonstrate its conclusion, lacked sufficient detail to be convincing. Additional discussions with the staff demonstrated that more complete analyses using the PRAISE code have led to the same conclusion. The PRAISE analyses of the failure probability of primary system piping assumed that a distribution of cracks existed in a component and calculated the probabilities of crack propagation through the wall

and failure. Parametric studies using the PRAISE code showed that the calculated probabilities of failure are small, even when very conservative loads and flaw-size distributions are assumed. The staff provided a careful quantification of uncertainty of fatigue crack initiation. We recommend such consideration of uncertainties in any future analyses regardless of the technical approach adopted.

We believe that the staff's conclusion concerning the risk significance of fatigue failure of reactor components is correct. Thus, we agree that this Generic Safety Issue is resolved.

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

Sincerely,

T. S. Kress

T. S. Kress
Chairman

References:
1. Memorandum dated August 18, 1995, from Charles Serpan, Jr., NRC Office of Nuclear Regulatory Research, to John T. Larkins, ACRS Executive Director, Subject: Proposed Resolution of Generic Safety Issue 78, "Monitoring of Fatigue Transient Limits for the Reactor Coolant System"
2. SECY-95-245 dated September 25, 1995, from James M. Taylor, Executive Director for Operations, to the Commissioners, Subject: Completion of the Fatigue Action Plan
3. Memorandum dated October 27, 1995, from Jeff Keisler and Omesh Chopra, Argonne National Laboratory, to Craig Hrabal, NRC Office of Nuclear Regulatory Research, Subject: Uncertainty Estimates for the Probability of Fatigue Crack Initiation in Reactor Components, NUREG/CR-6335, ANL-95/15
4. U. S. Nuclear Regulatory Commission, NUREG/CR-6237, "Statistical Analysis of Fatigue Strain-Life Data for Carbon and Low-Alloy Steels," August 1994
5. U. S. Nuclear Regulatory Commission, NUREG/CR-6335, "Fatigue Strain-Life Behavior of Carbon and Low-Alloy Steels, Austenitic Stainless Steels, and Alloy 600 in LWR Environments," June 1995

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

June 3, 1996

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

Dear Mr. Taylor:

SUBJECT: RESOLUTION OF THE MULTIPLE SYSTEM RESPONSES PROGRAM ISSUES

During the 431st meeting of the Advisory Committee on Reactor Safeguards, May 23-25, 1996, we completed our review of the adequacy of the resolution of the Multiple System Responses Program (MSRP) issues. During the 427th meeting, December 7-8, 1995, we heard presentations by and held discussions with representatives of the NRC staff and an ACRS Senior Fellow regarding this matter. We also had the benefit of the documents referenced.

In the process of reviewing a number of Unresolved Safety Issues (USIs) during the mid-1980s, the ACRS expressed concern that treating each safety issue in isolation might not identify significant system interactions. The ACRS also raised a number of questions concerning system interactions that were not addressed in the proposed resolution of certain USIs. Subsequently, the staff established the MSRP in 1986 to address ACRS concerns and other related issues.

The MSRP identified 21 potential generic issues. In August 1995, the NRC staff issued a final report which concluded that none of the MSRP issues posed new or separate safety concerns and that these issues were being addressed under the scope of the existing Generic Safety Issue (GSI) process, or in the programs of Individual Plant Examinations (IPEs) and Individual Plant Examination of External Events (IPEEEs).

The MSRP issues have been treated to a degree in the IPE/IPEEE programs and in the GSI process. A review of a number of IPE/IPEEE submittals, however, failed to identify satisfactory resolution for some issues (e.g., the treatment of interactions between nonsafety and safety systems, seismically induced interactions, and hydrogen line ruptures). We also note that the issues of nonsafety/safety systems interactions appear to be better treated in the IPEEE submittals that were based on probabilistic risk assessments than

in those that were based on Seismic Margins Methodology and Fire-Induced Vulnerability Evaluation Methodology.

Incorporation of some MSRP issues into the IPE/IPEEE process may have been expedient, but the staff failed to put into place a mechanism to ensure that licensees had evaluated and resolved these issues in an adequate manner. Additional staff review to determine the adequacy of the resolution of these issues is, therefore, warranted.

As stated in our report to the Commission, dated August 16, 1988, we continue to emphasize that "systems interactions, some of which may be adverse to safety, will continue to be revealed by operating experience in existing plants. These should be evaluated by the staff as they occur, and the lessons learned incorporated into the requirements and practices of the agency."

Sincerely,

T. S. Kress

T. S. Kress
Chairman

<u>References</u>:

1. U. S. Nuclear Regulatory Commission, NUREG/CR-5420, "Multiple System Responses Program - Identification of Concerns Related to a Number of Specific Regulatory Issues," Prepared by Oak Ridge National Laboratory, October 1989
2. Multiple System Responses Program - Final Report, transmitted by memorandum dated August 2, 1995 from L. C. Shao, Office of Nuclear Regulatory Research, to David L. Morrison, Office of Nuclear Regulatory Research
3. Memorandum dated January 12, 1996, from August W. Cronenberg, ACRS Senior Fellow, to ACRS Members and Staff, Subject: Observations from Review of Multiple System Responses Program (MSRP) Reports and Memoranda
4. U. S. Nuclear Regulatory Commission, NUREG-0933, "A Prioritization of Generic Safety Issues," July 1991
5. Report dated August 16, 1988, from W. Kerr, ACRS Chairman, to Lando W. Zech, Jr., NRC Chairman, Subject: Proposed Resolution of USI A-17, "Systems Interactions in Nuclear Power Plants"

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

December 13, 1996

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

Dear Mr. Taylor:

SUBJECT: ACRS REVIEW OF GENERIC LETTERS, BULLETINS, AND INFORMATION REQUESTS ISSUED ON AN EXPEDITED BASIS

During the 437th meeting of the Advisory Committee on Reactor Safeguards, December 5-7, 1996, we discussed our role in reviewing proposed generic letters, bulletins, and information requests issued pursuant to 10 CFR 50.54(f) by the staff on an expedited basis.

We are requesting that any generic letters, bulletins, and information requests issued pursuant to 10 CFR 50.54(f) be provided to the ACRS at the same time they are sent to the Committee to Review Generic Requirements. The ACRS Chairman and the cognizant Subcommittee Chairman will review these documents and inform the ACRS Executive Director of their decision with regard to the need for ACRS review before the documents are issued. The ACRS Executive Director will then expeditiously inform you of this decision.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

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

February 22, 1996

MEMORANDUM TO: James M. Taylor
Executive Director for Operations

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

SUBJECT: PROPOSED GENERIC COMMUNICATION REGARDING BORAFLEX DEGRADATION IN SPENT FUEL POOL STORAGE RACKS

During the 428th meeting of the Advisory Committee on Reactor Safeguards, February 8-10, 1996, the Committee decided not to review the proposed generic communication. The Committee appreciates being afforded the opportunity to review the subject matter.

Reference:
Memorandum dated November 2, 1995, from Dennis Crutchfield, Office of Nuclear Reactor Regulation, NRC, to David Meyer, Division of Freedom of Information and Publications Services, Office of Administration, NRC, Subject: Notice Of Opportunity For Public Comment for a Proposed Generic Communication Regarding BORAFLEX Degradation in Spent Fuel Pool Storage Racks (M19447)

cc: J. Hoyle, SECY
J. Blaha, OEDO
L. Soffer, OEDO
W. Russell, NRR
D. Crutchfield, NRR
A. Chaffee, NRR
J. Shapaker, NRR
L. Kopp, NRR
J. Cortez, RES

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

December 30, 1996

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

Dear Mr. Taylor:

SUBJECT: ACRS QUESTIONS ON HUMAN PERFORMANCE PROGRAM PLAN

During the 437th meeting of the Advisory Committee on Reactor Safeguards, December 5-7, 1996, we reviewed the NRC activities identified in the Human Performance Program Plan. Our Subcommittee on Human Factors met on September 20 and December 3, 1996, to review these activities. After the September 20, 1996 Subcommittee meeting, a list of questions included in the attachment was developed. These questions were provided to the staff on September 27, 1996. During subsequent meetings, the staff responded to these questions. We believe that the staff's response to questions 1, 2, 3, and 11, did not fully address our concerns. We request that the staff provide written response to these questions.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

Attachment: List of ACRS questions on Human Performance Program Plan

cc: J. Mitchell, OEDO
F. Miragila, NRR
B. Boger, NRR
C. Thomas, NRR
D. Morrison, RES
W. Hodges, RES

ATTACHMENT 1

LIST OF ACRS QUESTIONS ON HUMAN PERFORMANCE PROGRAM PLAN

The ACRS requested that the staff provide information at a future ACRS Subcommittee meeting concerning the following questions.

1. What are the staff plans for developing a Human Performance Program Plan (HPPP) activities road map, which would be useful for allocating resources, scheduling, and understanding the relationship between the activities?

2. The activities delineated in the HPPP appear to be focused on reducing the assumed risk-worth of human actions used in probabilistic risk assessments (PRA). What is the risk-worth of human actions? Why does the staff believe the risk-worth is too high and should be reduced?

3. How does the staff set the priorities for the HPPP activities and what does the priority ranking mean?

4. How does the staff decide that an independent program element is required? Why has the staff decided that data gathering should be separated from developing guidance and that the two activities should have different priorities?

[NOTE: The attached figures are examples of models that may be used to develop a master diagram that could serve as the road map to answer many of the questions raised here. These figures are just the starting point; they must be adapted to the NRC's needs using judgment and operational experience.]

5. What does the staff mean by "effective" and "adequate" as used in the objectives and goals in the HPPP? How does the staff know what must be done and when the goal or objective is achieved?

6. Should the staff be pushing licensees toward the state-of-the-art in human factors and human reliability rather than a proven adequate state?

7. Numerous human errors have resulted in the misadministration of medical treatments by licensees of the Office of Nuclear Materials Safety and Safeguards (NMSS). Why isn't NMSS as involved with human performance efforts as the other offices?

8. How does the staff plan to respond to the ACRS advice concerning developing metrics for organizations and managements that correlate with risk or performance?

9. What are the technical bases for defining the staffing levels inside and outside of the main control room, and for communication procedures?

10. What are the deficiencies or "holes" in NUREG-0700?

11. How are standards adopted by the staff formulated? How does the staff assure that the standards are necessary and sufficient to meet regulatory needs?

12. The staff scheduled item 1.2.11 of the HPPP, "Develop Guidance for Computerized Job Performance Aids," to be completed "as technology is developed." What standards does the staff have for such aids that would foster the development of such technology? If the standards do not exist, what are the staff plans for developing such standards?

13. What is the staff approach to developing a performance-based fitness-for-duty criteria?

14. What is the staff approach to evaluating the task network model espoused by the Department of Defense, and how will the staff decide if the model is applicable and useful for regulatory needs?

15. How does the staff decide on the allocation of resources between human factor research and other research activities such as thermal hydraulic models?

16. How does the staff assure simulator fidelity? How important is good fidelity to Emergency Operating Procedure training? What does the staff expect an operator to do if unexpected plant behavior occurs during a severe accident?

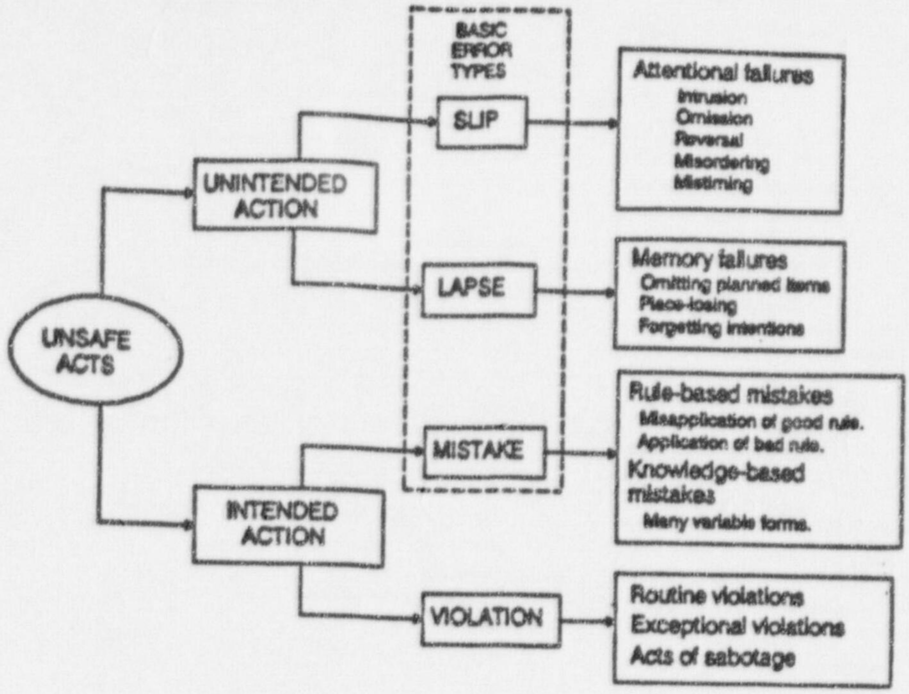

Figure 7.7. A summary of the psychological varieties of unsafe acts, classified initially according to whether the act was intended or unintended and then distinguishing errors from violations.

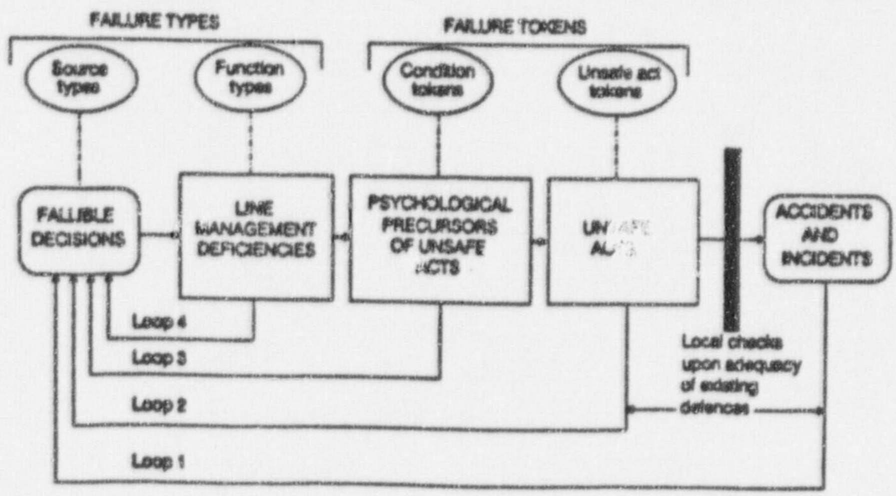

Figure 7.9. Feedback loops and indicators. The indicators are divided into two groups: *failure types* (relating to deficiencies in the managerial/organisational sectors) and *failure tokens* (relating to individual conditions and unsafe acts).

J. Reason, Human Error, Cambridge University Press, 1990.

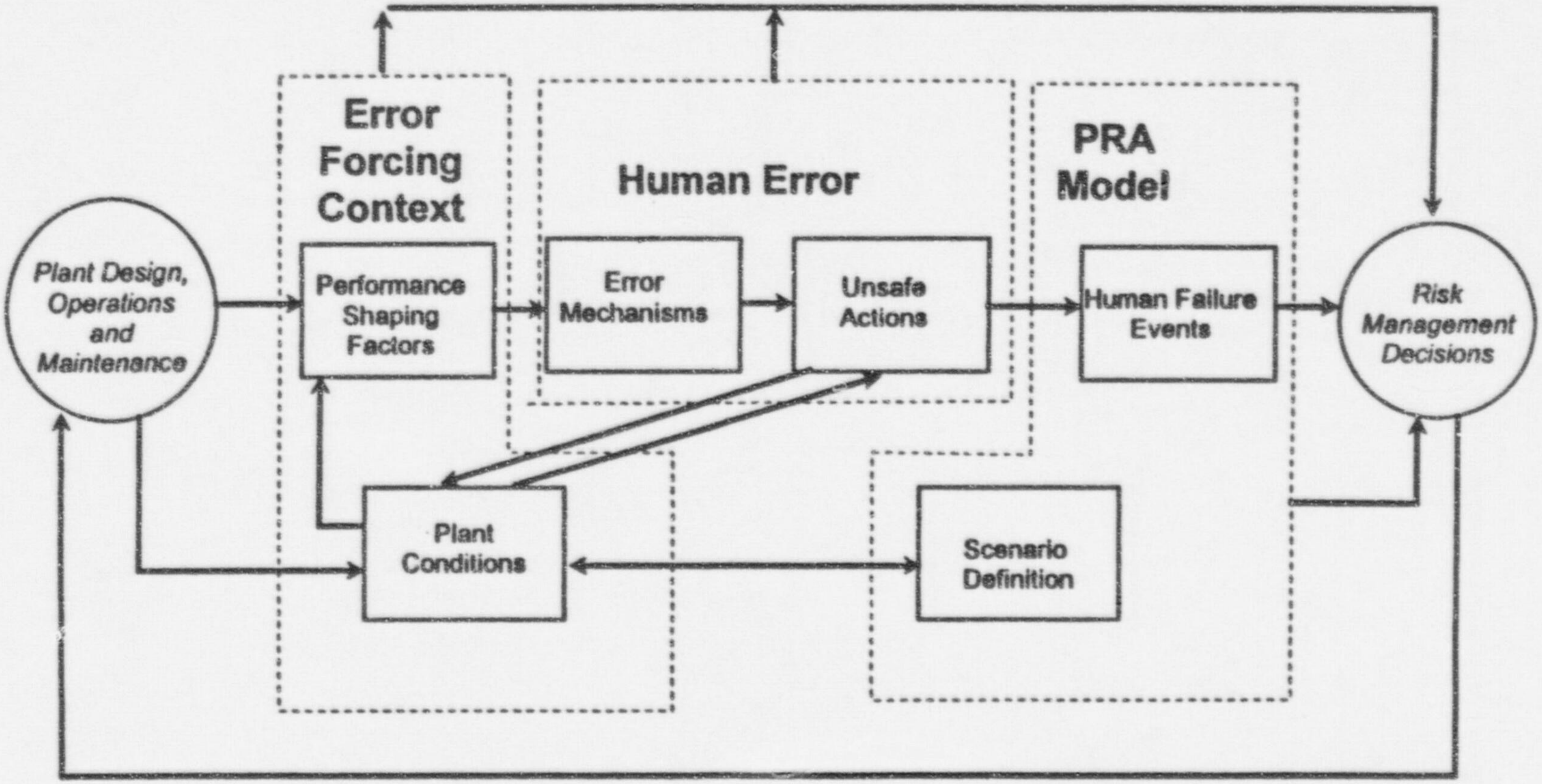

Figure 2.1 Multidisciplinary HRA framework

See *Figure 11.2, "Multifaceted taxonomy for description and analysis of events involving human malfunction," in* Information Processing and Human-Machine Interaction, (1986), by Jens Rasmussen, ISBN No. 0-444-00987-6

March 8, 1996

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

Dear Chairman Jackson:

SUBJECT: USE OF INDIVIDUAL PLANT EXAMINATIONS IN THE REGULATORY PROCESS

During the 428th and 429th meetings of the Advisory Committee on Reactor Safeguards, February 8-10 and March 7-9, 1996, respectively, we discussed the Individual Plant Examination (IPE) review process and findings with the NRC staff. Our Subcommittee on IPEs also met with the staff and its contractors on January 26, 1996, to review this matter. We also had the benefit of the documents referenced. This report is in response to the December 27, 1995 Staff Requirements Memorandum (SRM).

In the SRM, the Commission requested "the ACRS views on the extent to which the current spectrum of IPEs can be used in the regulatory process." We interpret this request as referring to potential regulatory uses of the IPEs that were not delineated in Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities." This report includes comments on both the Generic Letter goals and the Commission request.

Goals of Generic Letter 88-20

The purpose of the IPE program, as stated in Generic Letter 88-20, was for each licensee:

(1) to develop an appreciation of severe accident behavior

(2) to understand the most likely severe accident sequences that could occur at its plant

(3) to gain a more quantitative understanding of the overall probabilities of core damage and fission product releases

(4) to reduce, if necessary, the overall probabilities of core damage and fission product releases by modifying, where appropriate, hardware and procedures that would help prevent or mitigate severe accidents.

We note that the IPEs were to be limited to the examination of internal initiating events and internal floods with the reactor at power and that individual and societal risks were not to be estimated. Other programs deal with external events and shutdown risk.

The IPE program has been successful at most utilities in meeting goal (1) and, to a lesser extent, goals (2) and (3) of the Generic Letter. Goal (4) of the Generic Letter also appears to have been achieved. We were told that most licensees discovered weaknesses and took corrective actions. In addition, this program has been beneficial in educating a broader segment of the NRC staff about the issues related to these goals.

We were told by the staff that all licensees submitted a Level-1 probabilistic risk assessment (PRA). Most licensees also submitted a Level-2 PRA, although some addressed Level-2 phenomena in a rudimentary manner. The methods and data sources used by different licensees varied widely. In some cases, the choices appeared to be arbitrary. Some licensees chose to include common-cause failures only for major components, while others chose to ignore them completely.

It is difficult to determine the extent to which the variability in IPE results for similar classes of plants is due to actual plant differences or to modeling assumptions. Although some of the causes for this variability may be immediately apparent, others are not. The latter include assumptions made about success criteria, the assumed dependencies between operator actions, and the level of decomposition in fault-tree analyses. (We note that the fault trees were not requested as part of the IPE submittals.)

An example of a potentially significant impact of modeling differences is the range of core-damage frequencies (CDFs) for BWR 3/4s that the staff has compiled. This range is from about 10^{-7} to about 10^{-4} per reactor-year. Although the staff has stated that such differences are primarily due to plant differences, this range of results seems unrealistic given the similarity among BWR 3/4s.

Use of IPEs in the Regulatory Process

As discussed above, the quality and consistency of the IPEs vary and the impact of assumptions and analytical models is difficult to

assess. On a case-by-case basis, however, additional and extended use of these IPEs is possible. As specific regulatory issues arise, the PRA Standard Review Plan now being developed by the staff can serve as a template for judging the quality and acceptability of the individual plant PRA for the proposed application.

As the agency moves toward risk-informed regulation, there will be an increasing need for full-scope PRAs that incorporate fire risk, external events, other modes of operation, and site-specific consequences. When requests for risk-informed regulatory action arise, the NRC staff should make it clear that a relevant PRA should be used.

To achieve these goals, especially consistency, some degree of standardization will be required. Standardizing PRA models and methods has been a controversial subject. Proponents argue that it would create a basis for comparison of PRA results, while opponents fear that it would inhibit methodological developments. We recommend that IPEs be reviewed to identify acceptable and unacceptable assumptions and/or models. Codification of assumptions and models ought not inhibit the continued development of PRA methods. These activities would be a significant first step toward addressing the Commission's statement in the SRM dated June 16, 1995, "that more meaningful plant-to-plant or scenario-to-scenario comparisons based on risk could be achieved if PRAs were done on a more standardized, replicable basis."

We believe that the NRC could make additional use of the present IPEs (except those that the staff has found to use unacceptable methods or models) for a limited number of applications (e.g., regulatory analyses and prioritization of generic issues).

The staff stated that the CDFs for several PWRs are greater than 10^{-4} per reactor-year. Several BWRs have CDFs that are very close to 10^{-4} per reactor-year and the conditional containment failure probabilities for BWR Mark I containments range from about 0.02 to about 0.6. Although the PRAs have limitations as discussed above, these numbers suggest that an investigation would be warranted to reassess their validity and to verify that the very low numbers reported by some other plants reflect actual plant differences.

Our conclusion is that the IPE program has met successfully the objectives of Generic Letter 88-20. This program has developed a risk awareness, both in the utilities and the NRC, that will contribute significantly to efforts to establish a risk-informed and performance-oriented regulatory system. The plant-specific

IPEs are an extremely valuable asset that should not be permitted to languish unimproved and unused.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

References:

1. Staff Requirements Memorandum dated June 16, 1995, from Andrew L. Bates, Acting Secretary, NRC, to the File regarding Meeting with ACRS on June 8, 1995
2. Staff Requirements Memorandum dated December 27, 1995, from John C. Hoyle, Secretary, NRC, to John T. Larkins, ACRS regarding Meeting with ACRS on December 8, 1995
3. Generic Letter 88-20, dated November 23, 1988, to All Licensees Holding Operating Licenses and Construction Permits for Nuclear Power Reactor Facilities, Subject: Individual Plant Examination for Severe Accident Vulnerabilities - 10 CFR §50.54(f)

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

June 6, 1996

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

Dear Chairman Jackson:

SUBJECT: POTENTIAL USE OF IPE/IPEEE RESULTS TO COMPARE THE RISK OF THE CURRENT POPULATION OF PLANTS WITH THE SAFETY GOALS

This report is in response to a Staff Requirements Memorandum dated September 20, 1994, in which the Commission requested further guidance and insight on determining where the current population of operating plants, both individually and collectively, fall in relation to the safety goals. Our intent in developing a response was to examine the Individual Plant Examinations (IPEs)/Individual Plant Examinations of External Events (IPEEEs) results to see if they can be extended so as to compare the risk of the current population of plants with the safety goals.

During the 431st meeting of the Advisory Committee on Reactor Safeguards, May 23-25, 1996, we completed our discussions on this subject. During the 418th, February 1995, and 419th, March 1995 meetings, we heard presentations by an ACRS Senior Fellow on an approach for estimating the risk associated with some of the missing or incomplete elements of the IPEs. During our 431st meeting, we reviewed a study by the Brookhaven National Laboratory (BNL) (performed as part of the IPE Insights Program) that investigated the use of some of the IPEs to compare the plant risk to the safety goals. We also had the benefit of the documents referenced.

The prompt fatality and latent health effects quantitative safety goals are posed in risk terms. Consequently, to establish the status of the population of plants with respect to these goals, a full-scope Level 3 probabilistic risk assessment (PRA) of acceptable quality for every plant would seem to be required. Such PRAs would need to include all internal and external events (including low-power and shutdown operations) and would also need to take into consideration the individual site characteristics.

In almost all cases, the IPEs and IPEEEs are not and were not intended to be full-scope PRAs. For example, a large number of IPEEEs used the Fire Induced Vulnerability Evaluation (FIVE) Methodology to search for potential fire vulnerabilities and the Seismic Margins Methodology to search for seismic vulnerabilities, neither of which gives a direct

expression of risk. Furthermore, shutdown risk was not a part of the IPEs/IPEEEs. While most licensees performed some type of Level 2 containment analysis, the vast majority did not perform a Level 3 offsite consequences analysis.

The BNL study represents a good attempt to estimate the effects of some of the missing elements in the IPEs/IPEEEs. This study did not attempt to evaluate the risk resulting from seismic and fire events, nor did it attempt to evaluate risk in the shutdown mode.

Information is available that arguably would make it possible to bound the effects on risk of elements missing from the IPEs/IPEEEs and to develop an approximate comparison with the safety goals. Such a bound would be of questionable value and would have very large uncertainties. We do not recommend that this be done.

The evidence from the BNL study, NUREG-1150, other PRAs, and scoping studies of shutdown risk indicates that, on average, the population of plants meets the safety goals. A definitive determination of this, however, will only be possible when acceptable, full-scope Level 3 PRAs are available for all the plants. We believe that the required effort to develop such comprehensive PRAs cannot be justified for the sole purpose of comparison with the safety goals. Such PRAs, however, will be needed in the long run to move toward a coherent risk-informed regulatory system.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

<u>REFERENCES</u>:

1. Memorandum dated September 20, 1994, from John C. Hoyle, Acting Secretary, NRC, to John T. Larkins, Executive Director, ACRS, Subject: Staff Requirements - Periodic Meeting with ACRS, Thursday, September 8, 1994
2. Richard Sherry, ACRS Senior Fellow, "A Simplified Approach to Estimation of Seismic Core Damage Frequencies from a Seismic Margins Methods Analysis"
3. U. S. Nuclear Regulatory Commission, NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," Office of Nuclear Regulatory Research, December 1990
4. U. S. Nuclear Regulatory Commission, NUREG-XXXX, "Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance," Draft for Comment dated April 1996
5. U.S. Nuclear Regulatory Commission, NUREG/CR-6144, "Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Surry, Unit 1," Brookhaven National Laboratory, July 1994

6. U. S. Nuclear Regulatory Commission, NUREG/CR-6143, "Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Grand Gulf, Unit 1," Sandia National Laboratories, March 1995

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

June 6, 1996

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

Dear Mr. Taylor:

SUBJECT: REGULATORY GUIDANCE DOCUMENTS RELATED TO DIGITAL INSTRUMENTATION AND CONTROL SYSTEMS

During the 429th and 431st meetings of the Advisory Committee on Reactor Safeguards, March 7-9 and May 23-25, 1996, we reviewed portions of the proposed Standard Review Plan (SRP), Branch Technical Positions (BTPs), and Regulatory Guides related to digital instrumentation and control (I&C) systems. We held discussions with representatives of the NRC staff and its contractor, the Lawrence Livermore National Laboratory (LLNL). In addition, our Subcommittee on I&C Systems and Computers met with the NRC staff and LLNL to discuss these documents on March 6 and May 22, 1996. We also had the benefit of the documents referenced.

The staff requested ACRS to review the SRP Chapter 7 update in the early stages of development to accommodate the schedule set forth in the Digital I&C Task Action Plan. The staff expects to complete development of the SRP Chapter 7 update and associated guidance in September 1996, integrate the recommendations from the National Academy of Sciences/National Research Council (NAS/NRC) Phase 2 study report in October 1996, publish the Draft SRP Chapter 7 and associated guidance for public comment in December 1996, and issue the final SRP and related guidance in May 1997.

The staff is revising the SRP, adding two new sections, developing new BTPs, and preparing six regulatory guides that endorse eight industry standards. The staff presented a safety evaluation report (SER) on an Electric Power Research Institute (EPRI) topical report for electromagnetic/radiofrequency interference (EMI/RFI) design requirements and testing. A planned BTP on commercial off-the-shelf (COTS) software may be replaced by an SER on a topical report being developed by an EPRI working group. We concur with the staff conclusions in the SER associated with the EPRI topical report on EMI/RFI and encourage the staff to complete an SER for the EPRI topical report on COTS.

Considering the fact that the staff is using generally accepted U.S. software engineering practices, it appears that the staff approach is appropriate to update the SRP and associated guidance to codify the current regulatory framework for digital I&C. We raised several issues (e.g., the linkage between SRP Chapter 7 and other SRP chapters, and graded approaches based on importance to safety) that were subsequently clarified by the staff. The staff agreed to document these clarifications.

We have raised other issues that include the level of detail provided in the regulatory guides and the balance in the guidance between the review of the design process and the assessment of the product. We plan to report on these and other digital I&C issues at a later date.

We plan to review the staff's remaining SRP sections, the BTPs, and the SER on the EPRI topical report on COTS when they become available.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

References:

1. U. S. Nuclear Regulatory Commission, Standard Review Plan, Section 7.0, "Instrumentation and Controls—Overview of Review Process," Draft Version 3.0, February 12, 1996
2. U. S. Nuclear Regulatory Commission, Standard Review Plan, Section 7.1, "Instrumentation and Controls—Introduction," Draft Version 7.0, February 14, 1996
3. U. S. Nuclear Regulatory Commission, Standard Review Plan, Section 7.2, "Reactor Trip System," Draft Version 6.0, April 17, 1996
4. U. S. Nuclear Regulatory Commission, Standard Review Plan, Section 7.9, "Data Communications," Draft Version 4.1, April 18, 1996
5. U. S. Nuclear Regulatory Commission, (Proposed) Branch Technical Position HICB-14: "Guidance on Software Reviews for Digital Computer-Based Instrumentation and Control Safety Systems," Version 9.0, February 14, 1996
6. U. S. Nuclear Regulatory Commission, (Proposed) Branch Technical Position HICB-16: "Guidance on the Level of Detail Required for Design Certification Applications Under 10 CFR Part 52," Version 7.0, April 12, 1996

7. U. S. Nuclear Regulatory Commission, Draft Regulatory Guides, transmitted by memorandum dated February 9, 1996, from M. Wayne Hodges, Director, Office of Nuclear Regulatory Research, NRC, to John T. Larkins, ACRS:
   - U. S. Nuclear Regulatory Commission, Draft Regulatory Guide DG-XXXX, Version 2.7.2, "Verification, Validation, Reviews, and Audits for Digital Computer Software Used in Safety Systems of Nuclear Power Plants"
   - U. S. Nuclear Regulatory Commission, Draft Regulatory Guide DG-XXXX, Version 2.0.7, "Configuration Management Plans for Digital Computer Software Used in Safety Systems of Nuclear Power Plants"
8. U. S. Nuclear Regulatory Commission, Draft Regulatory Guides, transmitted by memorandum dated April 26, 1996, from M. Wayne Hodges, Director, Office of Nuclear Regulatory Research, NRC, to John T. Larkins, ACRS:
   - Draft Regulatory Guide DG-XXXX, Version 2.0, "Software Unit Testing for Digital Computer Software Used in Safety Systems of Nuclear Power Plants"
   - Draft Regulatory Guide DG-XXXX, Version 2.0, "Developing Software Life Cycle Processes for Digital Computer Software Used in Safety Systems of Nuclear Power Plants"
   - Draft Regulatory Guide DG-XXXX, Version 2.0, "Software Requirements Specifications for Digital Computer Software Used in Safety Systems of Nuclear Power Plants"
   - Draft Regulatory Guide DG-XXXX, Version 2.0, "Software Test Documentation for Digital Computer Software Used in Safety Systems of Nuclear Power Plants"
9. Memorandum dated January 30, 1996, from F. Miraglia, Office of Nuclear Reactor Regulation, NRC, to E. Jordan, Committee to Review Generic Requirements, NRC, Subject: Request for Endorsement of the Safety Evaluation Report on Electric Power Research Institute Topical Report, TR-102323, "Guidelines for Electromagnetic Interference Testing in Power Plants"

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

October 23, 1996

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

Dear Mr. Taylor:

SUBJECT: DRAFT UPDATE OF STANDARD REVIEW PLAN, CHAPTER 7, "INSTRUMENTATION AND CONTROLS"

During the 435th meeting of the Advisory Committee on Reactor Safeguards, October 9-12, 1996, we reviewed portions of the draft update of Standard Review Plan (SRP), Chapter 7, "Instrumentation and Controls." We heard presentations by and held discussions with representatives of the NRC staff and its contractor, the Lawrence Livermore National Laboratory (LLNL), regarding proposed SRP sections and Branch Technical Positions (BTPs) related to digital instrumentation and control (I&C) systems. In addition, our Subcommittee on Instrumentation and Control Systems and Computers met with the NRC staff and LLNL on October 8, 1996, to discuss this matter. We had previously met with the staff and LLNL in March and May 1996 to discuss draft SRP sections, BTPs, and associated regulatory guides, and provided comments in a letter dated June 6, 1996. We also had the benefit of the documents referenced.

We have no objection to the staff's proposal for issuing the draft update of SRP Chapter 7 and associated BTPs for public comment. However, in the June 6, 1996 letter, we identified issues regarding the level of detail provided in the regulatory guides, the balance in the guidance between the review of the design process and the assessment of the product, the linkage between Chapter 7 and other SRP chapters, and graded approaches based on importance to safety. In a letter dated June 21, 1996, you responded to our letter of June 6, 1996, stating that the staff will continue its discussions with the ACRS on these issues. We plan to discuss these matters with the staff during our future meetings.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

References:
1. Memorandum dated September 16, 1996, from Frank J. Miraglia, Jr., Office of Nuclear Reactor Regulation, to Edward L. Jordan, Committee to Review Generic Requirements, Subject: Request for Review of Updated Standard Review Plan Chapter 7, Instrumentation and Controls (attached)
2. Letter dated June 6, 1996, from T. S. Kress, ACRS Chairman, to James M. Taylor, Executive Director for Operations, Subject: Regulatory Guidance Documents Related to Digital Instrumentation and Control Systems
3. Letter dated June 21, 1996, from James M. Taylor, Executive Director for Operations, to T. S. Kress, ACRS Chairman, Subject: Regulatory Guidance Documents Related to Digital Instrumentation and Control Systems

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

November 20, 1996

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

Dear Mr. Taylor:

SUBJECT: PROPOSED RULE ON STEAM GENERATOR INTEGRITY

During the 436th meeting of the Advisory Committee on Reactor Safeguards, November 7-9, 1996, we reviewed the technical bases for the proposed steam generator integrity rule and an associated regulatory guide. During the 432nd meeting of the ACRS, June 12-14, 1996, and meetings of the Joint Subcommittees on Materials & Metallurgy and on Severe Accidents, June 3-4 and November 5-6, 1996, we heard presentations on subjects related to this matter. During these reviews, we had the benefit of discussions with representatives of the staff, the Nuclear Energy Institute, and the Electric Power Research Institute, as well as the author of a differing professional opinion. We also had the benefit of the documents referenced.

The proposed steam generator integrity rule is intended to provide a risk-informed and performance-based regulation to replace an existing prescriptive regulation. In its present form, the rule is a performance-based regulation almost completely divorced from any direct relation to risk objectives. Such a performance-based rule proliferates the incoherence problems of the present deterministic approach. The proposed rule preserves a tenuous connection between "design-basis space" and "risk space" without clearly articulating the risk objectives.

Some of the characteristics exhibited in the development process of the rule and regulatory guide include the following:

- difficulty in reaching agreement on the performance criteria,
- incomplete and sometimes perfunctory analyses required to provide an assessment of relative risk,
- reliance on core-damage frequency alone as an indicator of risk, and

- recourse to defense-in-depth without specific criteria for its use.

We believe that more direct consideration of risk could have avoided some of these difficulties.

A controversial element of the proposed rule and regulatory guide is the introduction of severe accident issues into an area that has been exclusively resolved by using a design-basis analysis. This extension of the scope of accident analysis is necessary to make risk-informed regulatory decisions and is part of the cost of moving toward risk-informed regulation. Since licensees have done risk-informed analyses for the Individual Plant Examination (IPE) process, we believe that the analysis for addressing severe accident events should not be overly burdensome to them.

Steam generator tube ruptures are small contributors to the total core-damage frequency, but may be risk significant due to containment bypass effects. In previous analyses, the staff performed limited assessments of primary side fission product attenuation and neglected secondary side attenuation. The regulatory guide now proposes that the licensees deal with the risk of a thermally induced tube failure either by demonstrating that the frequency of the initiating events is sufficiently low (10^{-6}/reactor year) or by demonstrating that the conditional probability of tube failure, given that an initiating event has occurred, is low (on the order of 0.1). We believe that licensees should also be given the option to demonstrate that, even if thermally induced tube ruptures occur, the associated risk is low when a more realistic treatment of fission product attenuation is made.

We are concerned that the proposed regulatory guide, as presented, could send the wrong message to licensees that risk-informed and performance-based requirements are add-ons to the traditional design-basis accident approach and can only result in an additional burden. We believe that to be risk informed and performance based, the regulatory guide should begin with a clear statement of its objectives, followed by a statement of the performance criteria and the guidelines for meeting the criteria. We note that the staff has stated that the proposed performance criteria have been derived from risk analyses, but we have not seen these analyses. Rewriting the regulatory guide is not a trivial task, but could result in a regulatory framework that could be used as a model for future risk-informed and performance-based rulemaking efforts.

In other applications of performance-based regulation such as the Maintenance Rule, the licensees have been permitted to determine appropriate performance criteria and have been given more flexibility in developing the methodology used to determine whether the criteria have been met. For the steam generator rule, the

staff has concluded that it should approve the performance criteria that are proposed by licensees to implement the steam generator rule. We agree with the decision of the staff that it should approve the criteria. Industry, however, should be provided more flexibility to propose alternative performance criteria supported by an appropriate risk analysis. We would like to review all of the supporting documentation before commenting on the specific criteria that have been proposed in the regulatory guide.

The demonstration that the criteria have actually been satisfied requires a complex process of nondestructive examination and evaluation of structural integrity and leakage during operation and design-basis accidents. The methodology required for these evaluations is not well established. Thus, the staff has felt constrained to provide a great deal of detail in the proposed regulatory guide to describe the characteristics of an acceptable methodology. Although we are not yet prepared to endorse the regulatory guide, we believe that the present immaturity of the methodology and the importance of the results justify such an approach.

The staff position is that the regulatory guide provides sufficient guidance for developing an acceptable methodology and that formal review of industry-developed repair criteria and procedures will not be required. We would like to review the results of a "trade study" of the preapproval approach vs. the post-implementation inspection approach to methodology acceptance.

Industry has questioned whether safety factors proposed in the steam generator rule are more conservative than those required by the ASME code. We encourage the staff to consider the industry's arguments.

Industry accepts the performance criterion proposed by the staff for primary-to-secondary leakage. Industry stated that this leakage criterion ought not be *ipso facto* a trigger for inspection or enforcement of regulations concerning the steam generator rule. This is a valid concern. Excessive leakage does not necessarily indicate a failure of the steam generator program. Adequate opportunities for staff action are available if failures of the program are discovered following a plant shutdown due to excessive primary-to-secondary leakage.

We are looking forward to reviewing the staff NUREG report concerning the staff's treatment of thermally induced tube failure. We are especially interested in the treatment of elevated temperatures resulting from flow through leaking tubes, and coupling between aerosol deposition and thermal hydraulics.

A differing professional opinion (DPO) was filed on July 11, 1994. We have reviewed the contentions in that DPO and summarized them in

the attachment. We also note that Generic Safety Issue (GSI)-163, "Multiple Steam Generator Tube Leakage," identified in 1992 has yet to be prioritized and resolved. Both the DPO and the GSI are directly related to the proposed rulemaking. We urge the staff to prepare a point-by-point response to the issues in the DPO and to prioritize and resolve GSI-163 before implementing the steam generator integrity rule.

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

Sincerely,

T. S. Kress

T. S. Kress
Chairman

Attachment:
Summary of Differing Professional Opinion Issues - Presented to the ACRS on November 7, 1996

<u>References</u>:

1. Memorandum dated October 25, 1996, from Brian Sheron, Office of Nuclear Reactor Regulation, to John Larkins, Executive Director, ACRS, Subject: ACRS Review of the Proposed Steam Generator Rule [forwarding the proposed steam generator rule and draft steam generator regulatory guide]
2. Memorandum dated May 1, 1996, from James M. Taylor, Executive Director for Operations, NRC, to Joram Hopenfeld, Office of Nuclear Regulatory Research, NRC, Subject: Resolution of Differing Professional Opinion Regarding Voltage-Based Repair Criteria for Steam Generator Tubes, dated July 13, 1994
3. Memorandum dated July 15, 1994, from James M. Taylor, Executive Director for Operations, NRC, to John T. Larkins, Executive Director, ACRS, Subject: ACRS Review Of Proposed Generic Letter 94-XX, Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes [forwarding Differing Professional Opinion]
4. 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"
5. Memorandum dated September 30, 1994, from Joram Hopenfeld, Office of Nuclear Regulatory Research, NRC, to John T. Larkins, Executive Director, ACRS, Subject: Comments On ACRS Review Of Generic Letter "Voltage Based Repair Criteria for Westinghouse Steam Generator Tubes"

SUMMARY OF DIFFERING PROFESSIONAL OPINION ISSUES
PRESENTED TO THE ACRS ON NOVEMBER 7, 1996

The DPO author estimates core-damage frequency with containment bypass to be 10^{-4} - 3.4×10^{-4} events/year. He stated that the uncertainties associated with characterizing steam generator tube defects and severe accident phenomena are not sufficiently understood to properly model tube rupture events. Tubes may fail before the surge line due to:

- crack networking and characterization of flaws not being adequately determined by nondestructive examinations,
- increased heat transfer caused by flow through tube cracks,
- cracks in tubes opening due to increased pressure,
- cracks in tubes unplugging at elevated pressure, and
- jets from tube cracks eroding adjacent tubes.

The DPO author stated that the staff should document the assumptions and models used to study hidden uncertainties.

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

April 23, 1996

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

Dear Chairman Jackson:

SUBJECT: PROBABILISTIC RISK ASSESSMENT FRAMEWORK, PILOT APPLICATIONS, AND NEXT STEPS TO EXPAND THE USE OF PRA IN THE REGULATORY DECISION-MAKING PROCESS

During the 430th meeting of the Advisory Committee on Reactor Safeguards, April 11-13, 1996, we continued our deliberations on risk-informed and performance-oriented regulation (RIPOR). We met with representatives of the NRC staff and the Nuclear Energy Institute (NEI) during our 429th meeting on March 7-9, 1996. Our Subcommittee on Probabilistic Risk Assessment (PRA) also met on October 26-27, 1995, with representatives of the NRC staff and of the nuclear industry, and on February 27-28, 1996, with the NRC staff and two invited experts, Dr. D. M. Karydas (performance-based standards for fire protection) and Professor T. G. Theofanous (on the proper formulation of safety goals and assessment of safety margins for rare and high-consequence hazards). We also had the benefit of the documents referenced.

This report is in response to the Staff Requirements Memorandum dated December 27, 1995, in which the Commission requested "ACRS views on the PRA framework document, its relationship to the pilot applications (SECY-95-280), and the next steps in the process to expand the use of PRA in the regulatory decision-making process."

PRA Framework Document

The PRA framework document provides a good starting point in the development of RIPOR. The six-step process described in the document is a reasonable way to proceed. We agree with the staff that the focus should be on the integration of probabilistic and deterministic approaches to regulation.

The PRA framework document, however, does not articulate an overall philosophy for RIPOR. We believe that such a philosophy should be developed. Some important high-level principles that should be included are:

1. RIPOR should consider risk from all modes of nuclear plant operations, including full power, shutdown, and transition.

2. The Commission's safety goals should serve as the top-level acceptance criteria.

3. Subsidiary performance-based acceptance criteria should be determined in a consistent way and must be measurable or calculable. The licensee should be granted flexibility in choosing the means to meet the criteria.

4. The relationship between RIPOR and defense-in-depth should be explained. The role of defense-in-depth in the determination of performance criteria to accommodate uncertainty and incompleteness in risk assessments should be established.

5. Criteria for the adoption of prescriptive regulations should be clearly delineated.

6. The acceptance criteria should be set at the highest level of plant system hierarchy that is consistent with the other principles noted above.

Discussion

It is indicative of the novelty of these concepts that we have spent a considerable amount of time discussing the meaning of "performance" among ourselves and with the staff and NEI. Some interpret performance in a limited way; i.e., its measures are simply the reliability and availability (or related quantities) of plant systems and components. Others take a broader view and interpret it as the overall performance of the licensee, including operations, maintenance, training, and the prevailing safety culture at the plant.

Similarly, the definition of performance criteria varies widely. At one extreme, we have simple measures that are either directly measurable or that involve calculations (e.g., the reliabilities and unavailabilities mentioned above). At the other extreme, performance criteria can be probabilistic or nonprobabilistic and can be set at any level. Observations and statistical or experimental evidence from the plant or other sources in conjunction with models can be used to demonstrate that the criteria have been met. As part of an overall philosophy, the staff needs to resolve the ambiguity in the definition of performance criteria.

Pilot Applications

While we support the staff's use of pilot applications, we are concerned that there seems to be no integrated justification for their selection. We would like to see the development of a list of important issues that are expected to arise on the road to RIPOR,

along with a discussion of how the selected pilot projects will help. The staff has agreed to look into these issues.

We also recommend that, for each pilot project, attempts be made to establish performance-based decision criteria along with the methods that would be used for demonstrating compliance. Such an exercise should provide useful insights regarding the overall feasibility of a performance-oriented approach to regulation.

Next Steps to Expand the Use of PRA in the Regulatory Decision-making Process

We believe that the NRC needs to take a number of important additional steps before a RIPOR environment can be achieved. These are discussed below.

Safety Goals

A restatement of the Commission's safety goal policy is needed that will allow the use of safety goals on a plant-specific basis.

Performance-Based Regulatory Criteria

A methodology is needed to determine performance-based criteria for regulatory action that are consistent with the top-level safety goals, as stated in the high-level principles. A "top-down" approach will ensure that this happens. An important element should be the preservation of the concept of defense-in-depth. The development of this methodology will also provide the opportunity to reexamine the validity of Level 2 subsidiary goals, which appear to be controversial at this time.

Programmatic Issues

Developing a RIPOR system should be a participative effort between the staff and the industry. We believe that the magnitude and significance of the task that the staff has undertaken requires a cooperative effort. Also, we recommend that the staff work with foreign researchers and regulatory agencies.

Conclusion

The intellectual and practical issues that the staff must confront in developing a RIPOR structure are significant. The staff has made a good start, but much remains to be done. We are pleased that the staff has agreed to meet with us periodically. Recent meetings have demonstrated that the staff is receptive to suggestions on how to deal with these complex issues. We applaud this attitude. We will keep you informed as these efforts progress.

Additional comments by ACRS Members Thomas S. Kress and Don W. Miller are presented below.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

Additional Comments by ACRS Members Thomas S. Kress and Don W. Miller

While we agree with most of the Committee's report on this subject, we find it to lack coherence. The major problem we have with the Committee report is its treatment of the concept of "performance-based" regulation. We conceive of basically two meanings to the word "performance" in this context: (1) the performance of equipment (systems and components) in carrying out the intended function, or (2) the performance of the licensee in performing its function (operation, maintenance, inspection, training, etc.). The first of these could further relate to either the operability of the specific equipment (e.g., does it turn on or off, and, in the case of a pump, for example, does it provide the required flow) or to the reliability/availability of the equipment. In our view, the former does not provide any basis on which to develop a regulatory structure (there are no meaningful acceptance criteria that relate to risk). On the other hand, the latter can clearly be anchored in risk. This, however, would be purely risk-based regulation. The word "performance" in this context becomes synonymous with "risk" and such a regulatory concept should be designated as risk-based and should not be called performance-based.

The second possible meaning of performance, the performance of the licensee, obviously has a nexus to risk. This connotation of performance, however, is what we have been calling organizational factors. To date, a methodology has not been developed by which objective performance measures can be identified and be factored directly into PRA to quantify risk implications. Therefore, at this time, we do not have the capability to develop such performance-based regulations in any coherent manner. This would, however, be an area worth pursuing in the future with additional research.

This leads us to our main point. At this time, we should be striving for risk-based or risk-informed regulations and should relegate the concept of "performance" regulation to being a remote possibility that needs substantial research to determine feasibility.

References:

1. Memorandum dated December 27, 1995, from J. Hoyle, Secretary of NRC, to J. Larkins, ACRS, Subject: Staff Requirements Memorandum dated December 27, 1995
2. Memorandum dated June 16, 1995, from A. Bates, Office of the Secretary, NRC, to File, Subject: Staff Requirements Memorandum dated June 16, 1995
3. Letter dated February 6, 1996, from J. Milhoan, Office of the Executive Director for Operations, NRC, to W. Rasin, Nuclear Energy Institute, Subject: Improving the Regulatory Process through Risk-Based and Performance-Based Regulation
4. Letter dated January 3, 1996, from J. Taylor, Executive Director for Operations, NRC, to Chairman Jackson, NRC, Subject: Improvements Associated With Managing the Utilization of Probabilistic Risk Assessment (PRA) and Digital Instrumentation and Control Technology
5. Letter dated November 30, 1995, from Chairman Jackson, NRC, to J. Taylor, Executive Director for Operations, NRC, Subject: Follow-up Requests in Probabilistic Risk Assessment and Digital Instrumentation and Control
6. SECY-95-280, "Framework for Applying Probabilistic Risk Analysis in Reactor Regulation," dated November 27, 1995
7. Letter dated November 14, 1995, from W. Rasin, Nuclear Energy Institute, to J. Milhoan, Office of Executive Director for Operations, NRC, Subject: Draft report, "Improving the Regulatory Process Through Risk-Based and Performance-Based Regulation"

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

August 15, 1996

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

Dear Chairman Jackson:

SUBJECT: RISK-INFORMED, PERFORMANCE-BASED REGULATION AND RELATED MATTERS

During the 433rd meeting of the Advisory Committee on Reactor Safeguards, August 8-10, 1996, we discussed the issues identified in the Staff Requirements Memorandum dated May 15, 1996. We also discussed the pilot applications for risk-informed, performance-based regulation. Our Subcommittee on Probabilistic Risk Assessment (PRA) met with representatives of the NRC staff and the nuclear industry on July 18 and August 7, 1996. We also had the benefit of the documents referenced.

The staff presentations dealt only with the development of guidelines from the Commission's safety goals to be used as an element of the evaluation of licensee-initiated changes to licensing commitments. All of our comments address the application of risk-informed regulation in that context. At a later time, we will discuss the larger question of the application of the safety goals on a plant-specific basis.

CONCLUSIONS

Issue 1: *Should the Commission's safety goals and subsidiary objectives be referenced or used to derive guidelines for plant-specific applications and, if so, how?*

We believe the safety goals and subsidiary objectives can and should be used to derive guidelines for plant-specific applications. It is, however, impractical to rely exclusively on the Quantitative Health Objectives (QHOs) for routine use on an individual plant basis. Criteria based on core damage frequency (CDF) and large, early release frequency (LERF) focus more sharply on safety issues and can provide assurance that the QHOs are met. They should be used in developing detailed guidelines.

<u>Issue 2</u>: *How are uncertainties to be accounted for?*

This is a difficult issue. There are models and formal methods to account explicitly for a large number of uncertainties. However, other uncertainties are unquantifiable. The staff proposes to explore a number of options, such as establishing margins in the acceptance guidelines, placing more importance on defense-in-depth, and others, to deal with such uncertainties. Such approaches seem appropriate, although much work remains to be done.

<u>Issue 3</u>: *Should requested changes to the current licensing basis be risk-neutral or should increases be permitted?*

We agree with the staff and industry that increases in risk should be permitted in some situations. Acceptance guidelines expressed in terms of the proposed change in risk and the current risk estimates should have three regions: a region in which some increase in risk is acceptable, one in which it is unacceptable, and one in which further analysis and evaluation would be required.

<u>Issue 4</u>: *How should performance-based regulation be implemented in the context of risk-informed regulation?*

We agree with the staff that, where practical, performance-based strategies should be included in the implementation and monitoring step of the risk-informed decision-making process. The pilot programs may provide an opportunity for a more concrete definition and development of performance-based strategies.

DISCUSSION

<u>Issue 1</u>

Even though a CDF could be derived from the QHOs that could be greater than 10^{-3} per reactor-year, the current subsidiary goal of 10^{-4} per reactor-year should be maintained and should be stated as a fundamental safety goal, along with the QHO. Accident sequences that have a high probability of leading to severe consequences could be controlled by the QHOs, but a more workable measure would be a subsidiary goal on the LERF. The definition of the latter needs to be improved. Whether the LERF should be a fixed value or derived from the QHOs, which would allow the LERF goal to include site-specific characteristics, needs to be investigated.

We recommend that the staff develop guidance for handling situations in which high values of the CDF occur for short periods of time (for example, 10^{-2} per reactor-year for a day).

Issue 2

In accounting for uncertainties, it is important to distinguish between those plant characteristics or phenomena that are modeled in the PRA and those that are not modeled (e.g., the actual layout of components and organizational factors). For those that are modeled, parameter and model uncertainties should be explicitly quantified and propagated through the PRA. The resulting distributions should be an input to the decision-making process along with other qualitative input.

Mean values of distributions should, in general, be used for comparison with goals or criteria, although the sensitivity of the mean value to the high tail of a distribution should not be overlooked. For very broad distributions, such as those that typically result when significant model uncertainty is present, reliance on the mean values may not be appropriate and a more detailed investigation of the reasons for this large uncertainty should be undertaken. This could possibly lead to decisions to conduct additional research or to take other measures.

Accounting for uncertainty in the case of plant characteristics or phenomena that are not currently modeled at all is much more difficult. The staff proposes to explore a number of options, such as establishing margins in the acceptance guidelines, placing more importance on defense-in-depth, and others. We agree and encourage the staff to actively pursue the resolution of this issue.

Issue 3

The concept of a "three-region" approach is consistent with the Electric Power Research Institute's PSA Applications Guide (PSAAG), although the boundaries of the regions used in the PSAAG are not necessarily the ones that the staff will adopt.

The staff has raised the issue of how "packaged" requests are to be handled. Packaging is the process by which risk trade-offs can be accomplished. It is a significant benefit of risk-informed regulation. We believe that it is the overall impact on plant risk that is important, and related changes should be handled as a package. Such changes should be consistent with the current philosophy of risk management; i.e., that the "bottom-line" numbers should not be the only input to the decision-making process, and other concepts such as defense-in-depth must be maintained.

We will continue to monitor the progress of the staff on these issues.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

References:
1. Staff Requirements Memorandum dated May 15, 1996, from John C. Hoyle, Secretary, NRC, to James M. Taylor, Executive Director for Operations, NRC, regarding Briefing on PRA Implementation Plan on April 4, 1996
2. Memorandum dated June 20, 1996, from James M. Taylor, Executive Director for Operations, NRC, to the Commission, Subject: Status Update of the Agency-Wide Implementation Plan for Probabilistic Risk Assessment (PRA) (from March 1, 1996 to May 31, 1996)
3. Electric Power Research Institute, EPRI TR-105396, Final Report dated August 1995, "PSA Applications Guide"

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

November 22, 1996

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

Dear Mr. Taylor:

SUBJECT: NRC PROGRAMS FOR RISK-BASED ANALYSIS OF REACTOR OPERATING EXPERIENCE

During the 436th meeting of the Advisory Committee on Reactor Safeguards, November 7-9, 1996, we reviewed the NRC programs for risk-based analysis of reactor operating experience. We heard presentations by and held discussions with representatives of the NRC staff regarding programs of the Office for Analysis and Evaluation of Operational Data (AEOD) including system reliability studies, risk-based performance indicators (PIs), accident sequence precursor (ASP) studies, and common-cause failures (CCFs). In addition, our Joint Subcommittees on Probabilistic Risk Assessment (PRA) and on Plant Operations met with representatives of the NRC staff and its contractors on July 17 and October 30, 1996, to review these matters. We also had the benefit of the documents referenced.

The AEOD staff presented a summary report of its programs for risk-based analysis of reactor operating experience. We found these programs to be comprehensive in covering the collection and analysis of operational safety data based on operating plant experience and balanced in providing results to both the immediate assessments for the NRC's plant PIs and the continuing longer range assembly of useful databases for system performance including CCF rates. We are convinced that careful review of operating experience is the most applicable source of information that the NRC and the industry have to validate system reliability analysis models and predictions, and is the best source of data for future use.

These databases have been developed through significant resource expenditures by the industry and the NRC. Both share the results of this effort through their independent analyses of event reports, system reliability data, etc. This information can be made useful only if the results are carefully reviewed for insights into system reliability, human performance, and utility and NRC management practices that may affect safety. The AEOD programs reflect an

awareness of the need to analyze these data intensively; however, the resources to perform a full scope analysis are not currently available. We urge that the priority assigned to this effort be revisited.

The NRC and the Institute of Nuclear Power Operations (INPO) have worked very hard to negotiate a more extensive sharing of their individual analysis products. These efforts have had some success, namely, NRC has gained access to data in the Nuclear Plant Reliability Data System of INPO, thus expanding the bases for NRC compilation of CCF data. Some concerns remain with regard to the protection of INPO proprietary rights. We believe any database used by NRC on CCF should be accessible to the public.

The CCF database that has been developed is a significant technical step forward. AEOD uses the database for generic evaluations. Plant-specific evaluation will almost certainly require modification to reflect configuration differences between the specific plant being considered and AEOD's generic evaluations. Provision should be made to caution any users of the CCF database of the limited applicability in its current form and, if possible, provide guidance on the proper process for modifying the database to reflect specific plant characteristics.

The AEOD staff presented some information on planned revisions to the NRC's PIs and initial efforts to incorporate risk-based PIs into the program. We look forward to further examination of candidate indicators. They must be carefully selected with a clear understanding of how the connection to risk is made and how this connection can be quantified. A first step will be the definition of the characteristics and attributes of risk-based PIs.

The AEOD staff is making progressive incremental improvements in its computational tools. It does not, however, have a long-range vision of the tools and resources that should be available to support risk-informed and performance-based regulation. We recommend that such a long-range plan be formulated for the development of computational tools.

The AEOD staff plans to enhance the ASP program to provide a more useful experience base for evaluating PRA results. The study of reliability of specific systems is a most important adjunct to

these studies. The planned addition to its study list of selected systems that are important to safety is timely. We welcome the opportunity to participate in this important work.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

References:
1. Office for Analysis and Evaluation of Operational Data report, "Risk-Based Analysis of Reactor Operating Experience," dated December 15, 1995
2. Memorandum dated March 22, 1996, from C. E. Rossi, Office for Analysis and Evaluation of Operational Data, NRC, to Office of Nuclear Reactor Regulation Directors and Regional Directors, NRC, Subject: Special Report - Emergency Diesel Generator Power System Reliability 1987-1993, INEL-95-0035 (1 volume)
3. Memorandum dated December 22, 1995, from C. E. Rossi, Office for Analysis and Evaluation of Operational Data, NRC, to G. Holahan, NRR, D. Crutchfield, M. Hodges and L. Shao, Office of Nuclear Regulatory Research, NRC, Subject: Common Cause Failure Parameter Estimates for Selected Components, INEL-94-0064 (6 volumes)

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

February 22, 1996

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

Dear Mr. Taylor:

SUBJECT: REVISION 2 TO REGULATORY GUIDE 1.149, "NUCLEAR POWER PLANT SIMULATION FACILITIES FOR USE IN OPERATOR LICENSE EXAMINATIONS"

During the 428th meeting of the Advisory Committee on Reactor Safeguards, February 8-10, 1996, we heard presentations by and held discussions with representatives of the NRC staff and the Nuclear Energy Institute concerning Revision 2 to Regulatory Guide 1.149. We also had the benefit of the documents referenced.

This revision to the Regulatory Guide describes a method acceptable to the NRC staff for complying with those portions of 10 CFR Part 55, "Operators' Licenses," that relate to the use of simulation facilities in the licensing of nuclear power plant operators. The current version of this Regulatory Guide endorses ANSI/ANS-3.5-1985, "Nuclear Power Plant Simulators for Use in Operator Training and Examinations," with some clarifications and exceptions. Revision 2 to the Regulatory Guide endorses ANSI/ANS-3.5-1993, again with some clarifications and exceptions. The NRC staff has met with industry representatives, including representatives of the ANSI/ANS-3.5 Working Group, to discuss the proposed Revision 2 to the Regulatory Guide and has considered industry comments in the proposed final version.

We believe that the staff should proceed with the publication of this Regulatory Guide to be consistent with the current state of the art with respect to the use of nuclear power plant simulators.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

References:
1. Memorandum dated January 30, 1996, from Bill M. Morris, Office of Nuclear Regulatory Research, NRC, to John T. Larkins, ACRS, Subject: Proposed Resolution of Draft Regulatory Guide DG-1043, Nuclear Power Plant Simulation Facilities for Use in Operator License Examinations
2. American Nuclear Society, ANSI/ANS-3.5-1993, "Nuclear Power Plant Simulators for Use in Operator Training and Examination," March 29, 1993

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

June 3, 1996

MEMORANDUM TO: James M. Taylor
Executive Director for Operations

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

SUBJECT: DRAFT REGULATORY GUIDE PERTAINING TO THE PREPARATION OF PETITIONS FOR RULEMAKING UNDER 10 CFR 2.802

During the 431st meeting of the Advisory Committee on Reactor Safeguards, May 23-25, 1996, the Committee decided not to review the subject draft regulatory guide. The Committee appreciates being afforded the opportunity to review the subject guide.

Reference:
Memorandum dated April 16, 1996, from David Morrison, RES, to John Larkins, ACRS, Subject: Draft Regulatory Guide Pertaining to the Preparation of Petitions for Rulemaking Under 10 CFR 2.802, and the Preparation and Submission of Proposals for Generic Regulatory Guidance Documents

cc: J. Hoyle, SECY
J. Blaha, OEDO
L. Soffer, OEDO
T. Martin, RES
D. Morrison, RES
J. Craig, RES
T. Chang, RES
J. Cortez, RES

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

June 12, 1996

MEMORANDUM TO: James M. Taylor
Executive Director for Operations

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

SUBJECT: REVISION 2 TO REGULATORY GUIDE 1.160, "MONITORING THE EFFECTIVENESS OF MAINTENANCE AT NUCLEAR POWER PLANTS"

During the 431st meeting of the Advisory Committee on Reactor Safeguards, May 23-25, 1996, the Committee considered the NRC staff request to waive the ACRS review and endorsement of Regulatory Guide 1.160 prior to issuing this Guide for industry use. Since the changes to Revision 2 to Regulatory Guide 1.160 were primarily clarifications, the Committee has no objection to issuing this Guide. The Committee, however, may wish to review the experience gained in implementing the provisions of this Guide sometime in the future.

Reference:
Memorandum dated May 9, 1996 from Ashok C. Thadani, NRR, to T. S. Kress, Chairman, ACRS, Subject: Expedited Issuance for Revision 2 to Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants"

cc: J. Hoyle, SECY
J. Blaha, OEDO
L. Soffer, OEDO
D. Morrison, RES
J. Cortez, RES
F. Kantor, NRR
A. Thadani, NRR

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

June 18, 1996

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

Dear Mr. Taylor:

SUBJECT: DRAFT REGULATORY GUIDE DG-1047, "STANDARD FORMAT AND CONTENT FOR APPLICATIONS TO RENEW NUCLEAR POWER PLANT OPERATING LICENSES"

During the 432nd meeting of the Advisory Committee on Reactor Safeguards, June 12-14, 1996, we discussed the subject draft Regulatory Guide with representatives of the NRC staff and the Nuclear Energy Institute. We also had the benefit of the documents referenced.

We have no objection to the staff proposal to issue the draft Regulatory Guide for public comment. We plan to review the proposed final version of this Guide after reconciliation of the public comments.

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. U.S. Nuclear Regulatory Commission, Draft Regulatory Guide DG-1047, "Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses," transmitted by memorandum dated April 18, 1996, from Scott F. Newberry, Office of Nuclear Reactor Regulation, NRC, to John T. Larkins, ACRS
2. Nuclear Energy Institute, NEI 95-10 (Revision 0), "Industry Guideline for Implementing the Requirements of 10 CFR Part 54—The License Renewal Rule," March 1996

3. U.S. Nuclear Regulatory Commission, SECY-96-059 dated March 18, 1996, from James M. Taylor, Executive Director for Operations, NRC, for the Commissioners, "Activities Associated with the Implementation of 10 CFR Part 54"

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

August 13, 1996

MEMORANDUM TO: James M. Taylor
Executive Director for Operations

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

SUBJECT: PROPOSED REVISION 3 TO REGULATORY GUIDE 1.8, "QUALIFICATION AND TRAINING OF PERSONNEL FOR NUCLEAR POWER PLANTS"

During the 433rd meeting of the Advisory Committee on Reactor Safeguards, August 8-10, 1996, the Committee considered the NRC staff request to issue proposed Revision 3 to Regulatory Guide 1.8 for public comment. The Committee has no objection to the issuance of this proposed Regulatory Guide for public comment. The Committee plans to review the proposed final version of this Regulatory Guide after reconciliation of public comments.

Reference:
Memorandum dated June 20, 1996, from M. Wayne Hodges, RES, to John T. Larkins, ACRS, Subject: Issuance of Regulatory Guide 1.8, Revision 3, for Public Comment without Prior ACRS Review

cc: J. Hoyle, SECY
J. Blaha, OEDO
L. Soffer, OEDO
W. Russell, NRR
W. Hodges, RES
F. Coffman, RES
J. Cortez, RES

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

August 14, 1996

MEMORANDUM TO: James M. Taylor
Executive Director for Operations

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

SUBJECT: PROPOSED REVISIONS TO REGULATORY GUIDES 1.84, 1.85, AND 1.147 PERTAINING TO ASME CODE CASES

During the 433rd meeting of the Advisory Committee on Reactor Safeguards, August 8-10, 1996, the Committee decided not to review the proposed revisions to Regulatory Guides 1.84, "Design and Fabrication Code Case Acceptability, ASME Section III, Division 1;" 1.85, "Materials Code Case Acceptability, ASME Section III, Division 1;" and 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1." The Committee would like to have the opportunity to review future regulatory guides that pertain to ASME Code Cases.

Reference:
Memorandum dated July 11, 1996, from Edward O. Woolridge, RES, to John T. Larkins, ACRS, Subject: Regulatory Guide Review

cc: J. Hoyle, SECY
J. Blaha, OEDO
L. Soffer, OEDO
D. Morrison, RES
L. Shao, RES
M. Mayfield, RES
E. Woolridge, RES
J. Cortez, RES
G. Mizuno, OGC

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

August 16, 1996

MEMORANDUM TO: James M. Taylor
Executive Director for Operations

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

B. Savio for

SUBJECT: PROPOSED REVISION 3 TO THE REGULATORY GUIDE 1.105, "INSTRUMENT SETPOINTS FOR SAFETY SYSTEMS" (DRAFT REGULATORY GUIDE DG-1045)

During the 433rd meeting of the Advisory Committee on Reactor Safeguards (ACRS), August 8-10, 1996, the Committee considered the NRC staff request for ACRS review of proposed Regulatory Guide 1.105, Revision 3. We have no objection to the issuance of the proposed Regulatory Guide for public comment. The Committee plans to review the proposed final version of this Regulatory Guide after reconciliation of public comments.

Reference:

Memorandum dated July 10, 1996, from Lawrence C. Shao, RES, to John T. Larkins, ACRS, Subject: Proposed Revision 3 to Regulatory Guide 1.105, "Instrument Setpoints for Safety Systems" (Draft Regulatory Guide DG-1045)

cc: J. Hoyle, SECY
J. Blaha, OEDO
L. Soffer, OEDO
W. Russell, NRR
L. Shao, RES
M. Mayfield, RES
J. Cortez, RES
S. Aggarwal, RES

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

November 18, 1996

MEMORANDUM TO: James M. Taylor
Executive Director for Operations

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

SUBJECT: PROPOSED FINAL REGULATORY GUIDE PERTAINING TO THE PREPARATION OF PETITIONS FOR RULEMAKING UNDER 10 CFR 2.802

During the 436th meeting of the Advisory Committee on Reactor Safeguards, November 7-9, 1996, the Committee decided not to review the subject regulatory guide.

Reference:
Memorandum dated October 22, 1996, from D. Morrison, Director, Office of Nuclear Regulatory Research, to J. Larkins, Executive Director, ACRS, Subject: Regulatory Guide entitled "Petitions for Rulemaking Under 10 CFR 2.802 and the Preparation and Submission of Proposals for Generic Regulatory Guidance Documents"

cc: J. Hoyle, SECY
J. Blaha, OEDO
J. Mitchell, OEDO
D. Morrison, RES
J. Craig, RES
T. Chang, RES
J. Cortez, RES

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

March 14, 1996

MEMORANDUM TO: James M. Taylor
Executive Director for Operations

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

SUBJECT: PROPOSED REVISION 8 TO NUREG-1021, "OPERATOR LICENSING EXAMINATION STANDARDS FOR POWER REACTORS"

During the 429th meeting of the Advisory Committee on Reactor Safeguards, March 7-9, 1996, the Committee decided not to review the proposed Revision 8 to NUREG-1021. The Committee appreciates being afforded the opportunity to review the subject matter.

Reference:
Proposed Revision 8 to NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," February 1996

cc: J. Hoyle, SECY
J. Blaha, OEDO
L. Soffer, OEDO
W. Russell, NRR
B. Boger, NRR
S. Richards, NRR
S. Guenther, NRR
J. Cortez, RES

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

April 17, 1996

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

Dear Chairman Jackson:

SUBJECT: CONTINUED NEED FOR UNITED STATES MEMBERSHIP IN THE NUCLEAR ENERGY AGENCY

The Advisory Committee on Reactor Safeguards has recently learned of the proposed withdrawal of the United States (U.S.) from participation in the Nuclear Energy Agency (NEA), a part of the Organization for Economic Cooperation and Development. The Department of Energy (DOE) and the Nuclear Regulatory Commission (NRC) are the primary U.S. technical participants in the NEA activities and, hence, are the agencies that have the most complete understanding of the benefits of membership in NEA. Our comments will perhaps assist you as you set forth the NRC position.

For many years, the NEA Committee on Safety of Nuclear Installations (CSNI) has been an active and productive leader in nuclear reactor safety research. CSNI reports cover the full scope of reactor safety concerns and are prepared by leading technical experts from the primary technical research laboratories and agencies of the member countries.

Current CSNI efforts contribute to U.S. programs in extended burnup reactor fuels, high-pressure melt ejection, direct containment heating, accident management, and steam explosions. Clearly, the CSNI has demonstrated the ability to keep pace with real concerns in nuclear safety. Furthermore, these efforts have resulted in substantial savings in U.S. research costs.

Nuclear safety is truly an international concern. The NEA is a forum for the consideration of common technical safety issues by the responsible regulatory agencies in the member countries and has been useful in developing consistent "western" positions. If the NEA did not exist, we would soon be convinced that it should be invented.

We believe that the suggested U.S. withdrawal from the NEA is shortsighted. We fully support your efforts to ensure continued U.S. participation in the NEA.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

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

OFFICE OF
ACRS/ACNW

April 22, 1996

MEMORANDUM TO: James M. Taylor
Executive Director for Operations

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

SUBJECT: PROPOSED STANDARD REVIEW PLAN FOR DRY CASK STORAGE SYSTEMS

Based on the recommendations of the Joint Subcommittee of the Advisory Committee on Reactor Safeguards (ACRS) and Advisory Committee on Nuclear Waste (ACNW), which met on March 26, 1996, the ACRS and ACNW decided not to review the subject proposed Standard Review Plan at this time. The ACRS and ACNW appreciate being afforded the opportunity to review the subject matter.

Reference:
NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems," dated February 1996

cc: J. Hoyle, SECY
J. Blaha, OEDO
L. Soffer, OEDO
W. Travers, NMSS
C. Haughney, NMSS
J. Cortez, RES

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

May 31, 1996

MEMORANDUM TO: James M. Taylor
Executive Director for Operations

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

SUBJECT: ISSUANCE OF THE UPDATED STANDARD REVIEW PLAN FOR PUBLIC COMMENT WITHOUT ACRS REVIEW

During the 431st meeting of the Advisory Committee on Reactor Safeguards, May 23-25, 1996, the Committee considered the NRC staff request to issue the updated Standard Review Plan (SRP) for public comment without ACRS review. The Committee has no objection to the issuance of the proposed SRP update for public comment. The Committee may, however, wish to review portions of the proposed final SRP after reconciliation of public comments.

Reference:
Memorandum dated April 9, 1996, from Frank Miraglia, NRR, to John Larkins, ACRS, Subject: Issuance of the Updated Standard Review Plan for Public Comment Without Prior ACRS Review

cc: J. Hoyle, SECY
J. Blaha, OEDO
L. Soffer, OEDO
W. Russell, NRR
F. Miraglia, NRR
F. Gillespie, NRR
D. Morrison, RES
J. Cortez, RES

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

June 5, 1996

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

Dear Mr. Taylor:

SUBJECT: IMPLEMENTATION OF THE REGULATORY REVIEW GROUP RECOMMENDATIONS

During the 431st meeting of the Advisory Committee on Reactor Safeguards, May 23-25, 1996, we reviewed the status of the implementation of the Regulatory Review Group recommendations. During our review, we had the benefit of discussions with representatives of the NRC staff and the referenced document.

The Regulatory Review Group was established by you on January 4, 1993, to conduct a comprehensive and disciplined review of power reactor regulations and related NRC procedures, programs, and practices. In August 1993, the Regulatory Review Group issued its final report containing recommendations to reduce the regulatory burden on licensees and to strengthen NRC administrative practices. The staff submitted its plan for implementing these recommendations in January 1994 and issued subsequent semiannual status reports.

We believe that the effort by the Regulatory Review Group has been successful. The Regulatory Review Group recommendations have been implemented or assigned to appropriate NRC offices for implementation. We would like to compliment the staff on its success.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

<u>Reference</u>:
SECY-96-024, dated February 2, 1996, from James M. Taylor, Executive Director for Operations, NRC, for the Commissioners, Subject: Semiannual Status Report on the Implementation of Regulatory Review Group Recommendations

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

August 14, 1996

MEMORANDUM TO: James M. Taylor
Executive Director for Operations

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

SUBJECT: DRAFT STANDARD REVIEW PLANS ON ANTITRUST AND POWER REACTOR LICENSEE FINANCIAL QUALIFICATIONS & DECOMMISSIONING FUNDING ASSURANCE

During the 433rd meeting of the Advisory Committee on Reactor Safeguards, August 8-10, 1996, the Committee decided not to review the subject standard review plans. The Committee appreciates the opportunity to review the subject matter.

Reference:
Memorandum dated June 27, 1996, from David Matthews, Office of Nuclear Reactor Regulation, to Lawrence Chandler, Office of the General Counsel, Subject: Standard Review Plans on Antitrust and Financial Qualifications & Decommissioning Funding Assurance

cc: J. Hoyle, SECY
J. Blaha, OEDO
L. Soffer, OEDO
W. Russell, NRR
B. Grimes, NRR
D. Matthews, NRR
J. Cortez, RES
L. Chandler, OGC

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

February 29, 1996

MEMORANDUM TO: James M. Taylor
Executive Director for Operations

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

SUBJECT: PROPOSED FINAL AMENDMENT TO 10 CFR PART 50 REGARDING FREQUENCY OF EMERGENCY PLANNING EXERCISES AT NUCLEAR POWER PLANTS

During the 428th meeting of the Advisory Committee on Reactor Safeguards, February 8-10, 1996, the Committee decided not to review the proposed final amendment. The Committee appreciates being afforded the opportunity to review the subject matter.

Reference:
Memorandum dated January 30, 1996, from David Morrison, Director, RES, to Edward Jordan, Director, AEOD; William Russell, Director, NRR; et. al., Subject: Office Review and Concurrence on Final Amendments to 10 CFR Part 50 Regarding Frequency of Emergency Planning Exercises at Nuclear Power Plants

cc: J. Hoyle, SECY
J. Blaha, OEDO
L. Soffer, OEDO
D. Morrison, RES
B. Morris, RES
T. Martin, RES
M. Jamochian, RES
J. Cortez, RES

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

April 22, 1996

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

Dear Chairman Jackson:

SUBJECT: PROPOSED REVISIONS TO 10 CFR PARTS 50 AND 100 AND PROPOSED REGULATORY GUIDES RELATING TO REACTOR SITE CRITERIA

During the 430th meeting of the Advisory Committee on Reactor Safeguards, April 11-13, 1996, we reviewed the proposed revisions to reactor siting regulations and associated Regulatory Guides and Standard Review Plan sections. Our Subcommittee on Extreme External Phenomena reviewed this matter during a meeting on April 3, 1996. During this review, we had the benefit of discussions with representatives of the NRC staff, Westinghouse Electric Corporation, and the Nuclear Energy Institute. We also had the benefit of the document referenced.

The staff has proposed final revisions to 10 CFR Parts 50 and 100 and a new Appendix S to Part 50 that deal with both seismic and source term issues for future plants and sites. Many of the implementation details will be found in new Regulatory Guides and in Standard Review Plan sections. The existing requirements of 10 CFR Part 100 and its Appendix A will remain in effect for operating plants.

We recommend that the proposed final rule dealing with the seismic aspects be issued.

The proposed final rule requires that any individual, located at any point on the exclusion area boundary for any two-hour period following the postulated release of the fission products, not receive a radiation dose in excess of 25 rem total effective dose equivalent (TEDE). Similarly, an individual located at the outer boundary of the low population zone (LPZ), who is exposed to the radioactive cloud resulting from the release of the postulated fission products (during the entire period of its passage), not receive a dose in excess of 25 rem TEDE. Consistency within the body of NRC regulations is most desirable. We recommend that careful definitions of the TEDE limits that are mindful of organ dose weighting factors found in 10 CFR Part 20 be included in the final rule.

Radiological doses are to be evaluated over a two-hour period. The proposed final rule states that the evaluation should be over the two-hour period of maximum dose. The Office of Nuclear Regulatory Research (RES) has a differing view and recommends that the proposed final rule be modified from any two-hour period after release of fission products (referred to as the "worst" two hours) to a period of two hours commencing with fuel failure (referred to as the "first" two hours). RES believes that the use of the worst two-hour period in the dose calculation is not justified by risk considerations and could lead to increased costs for future licensees with no commensurate gain in safety.

The staff supporting the proposed rule states that (1) the proposed licensing framework would provide a relaxation of engineered safety feature (ESF) performance requirements commensurate with updated source term and radiological insights, (2) the regulatory requirements for determination of in-containment radioactive material during the two-hour dose evaluation period would be consistent and capable of handling designs substantially different from those analyzed in NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," (3) the analysis would be easy to perform and reproducible with confidence, and (4) the technical bases and analytical methods would be defensible. While the revised dose evaluation in 10 CFR 50.34 is intended for future plants, the staff is concerned that a current licensee might seek to use it to remove or disable existing fission product cleanup systems. This could markedly change the risk profile of the plant from that which was licensed.

We are not persuaded by the rationale provided by RES in favor of the first two-hour dose calculation. We agree with the position taken in the proposed final rule, and recommend that the rule and the associated Regulatory Guides and SRP sections be issued.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

<u>REFERENCE</u>:
Memorandum dated March 6, 1996, from T. P. Speis, Office of Nuclear Regulatory Research, NRC, to J. T. Larkins, ACRS, transmitting Revisions to 10 CFR Part 100, Reactor Site Criteria, Revisions to 10 CFR Part 50, New Appendix S to Part 50 (Final Rules) and Associated Regulatory Guides and Standard Review Plan Sections

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

June 4, 1996

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

Dear Mr. Taylor:

SUBJECT: PROPOSED RULE ON SHUTDOWN OPERATIONS

During the 431st meeting of the Advisory Committee on Reactor Safeguards, May 23-25, 1996, we held discussions with representatives of the NRC staff and the Nuclear Energy Institute (NEI), concerning the subject proposed rule and the probabilistic risk assessment (PRA) studies that were performed for the Surry and the Grand Gulf Nuclear Power plants. Our Subcommittee on Plant Operations met with the staff, NEI, and a utility representative on May 21, 1996, to discuss these matters. We also had the benefit of the documents referenced. We previously commented on the staff effort to resolve the shutdown operations issue in our letters dated August 13, 1991, April 9, 1992, September 15, 1992, and May 13, 1994.

According to the staff, the proposed rule will contain performance-based elements. Since the supporting regulatory analysis and regulatory guide are still being developed, we discussed only the proposed rule during our meeting. The staff has held several public meetings with NEI to obtain industry input on the formulation of this rule.

We made a number of comments on the risk basis for the rule. The staff agreed to consider our comments as it finalizes the draft rule, which it plans to publish for public comment in September 1996. We plan to provide comments on the proposed final rule after the staff has reconciled the public comments.

The concern for risk associated with shutdown operations has arisen from incidents that have occurred. Our quantitative understanding of the risk posed by plants in low-power or shutdown modes of operation is limited. Risk assessments for shutdown operations were performed for Surry (a three-loop PWR with loop isolation valves and a sub-atmospheric pressure containment) and Grand Gulf (a BWR-6 with a Mark III containment). Neither of these plants is a particularly good surrogate for the entire population of PWRs and BWRs.

The studies of shutdown risk consisted of two phases. The first phase was a deliberately conservative scoping analysis. The second phase focused on a single, high-risk plant operational state among the many that exist during shutdown operation. Such an approach could lead to an incorrect assessment of risk (a historical analogue is the selection of the large-break, loss-of-coolant accident as a bounding event) or to the adoption of operating practices that might increase risk.

The available evidence does suggest that shutdown operations can make important contributions to the overall risk to the public posed by nuclear power plants. On the eve of our entry into an era of risk-informed rulemaking, there are no complete, reliable assessments of risk during shutdown operations even for a few representative plants. Certainly, there is nothing commensurate with the NUREG-1150 study of risk during full-power operation.

The staff effort toward an interim solution by promulgating this proposed rule is based on engineering judgment and will probably lessen risk. A risk-informed understanding will require a quantitative evaluation of risk during low-power and shutdown operations. We therefore recommend that priority attention be given to performing Level 3 PRAs for shutdown operations at the NUREG-1150 plants with consideration of spent fuel pool risk and uncertainty assessments.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

References:
1. Memorandum dated April 5, 1996, from Robert C. Jones, Office of Nuclear Reactor Regulation, to John T. Larkins, ACRS, Subject: Development of §50.67, "Shutdown Operation of Nuclear Power Plants"
2. U. S. Nuclear Regulatory Commission, Prepared by Brookhaven National Laboratory, NUREG/CR-6144, "Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Surry, Unit 1," Summary of Results, October 1995
3. U. S. Nuclear Regulatory Commission, Prepared by Sandia National Laboratories, NUREG/CR-6143, "Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Grand Gulf, Unit 1," Summary of Results, July 1995
4. Nuclear Management and Resources Council, Inc., NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management," December 1991

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

August 14, 1996

MEMORANDUM TO: James M. Taylor
Executive Director for Operations

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

SUBJECT: RULEMAKING PLAN FOR AMENDMENTS TO 10 CFR 73.55, CHANGES TO NUCLEAR POWER PLANT SECURITY REQUIREMENTS

During the 433rd meeting of the Advisory Committee on Reactor Safeguards, August 8-10, 1996, the Committee decided not to review the subject rulemaking plan. The Committee appreciates the opportunity to review this subject matter.

Reference:
SECY-96-105, dated May 14, 1996, from James M. Taylor, Executive Director for Operations, for the Commissioners, Subject: Rulemaking Plan for Amendments to 10 CFR 73.55, Changes to Nuclear Power Plant Security Requirements

cc: J. Hoyle, SECY
J. Blaha, OEDO
L. Soffer, OEDO
D. Morrison, RES
J. Cortez, RES
W. Russell, NRR

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

February 27, 1996

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

Dear Mr. President:

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

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

Sincerely,

T. S. Kress

T. S. Kress
Chairman

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

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

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* For Items 1 through 15, see NUREG-1125, Volume 17, 4/96.

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

February 27, 1996

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

Dear Mr. Speaker:

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

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

Sincerely,

T. S. Kress

T. S. Kress
Chairman

Enclosures:

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

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7. Report from T. S. Kress, ACRS Chairman, to James M. Taylor, NRC Executive Director for Operations, Subject: Proposed Commission Paper on Staff Positions on Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design, June 15, 1995
8. Report from T. S. Kress, ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: Proposed Final Policy Statement on the Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities, June 16, 1995
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10. Report from T. S. Kress, ACRS Chairman, to James M. Taylor, NRC Executive Director for Operations, Subject: The Nuclear Energy Institute Petition for Rulemaking to Amend 10 CFR 50.48, "Fire Protection," September 15, 1995
11. Report from T. S. Kress, ACRS Chairman, to Shirley A. Jackson, NRC Chairman, Subject: Development of Improved Nondestructive Examination (NDE) Techniques, September 15, 1995
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15. Report from T. S. Kress, ACRS Chairman, to James M. Taylor, NRC Executive Director for Operations, Subject: Proposed Final Regulatory Guide 1.164, "Time Response Design Criteria for Safety-Related Operator Actions," to Resolve Generic Safety Issue B-17, November 14, 1995

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* For Items 1 through 15, see NUREG-1125, Volume 17, 4/96.

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

June 28, 1996

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

Dear Chairman Jackson:

SUBJECT: SEVERE ACCIDENT RESEARCH

During the 432nd meeting of the Advisory Committee on Reactor Safeguards, April 11-13, 1996, we completed our review of the status of the NRC severe accident research program and severe accident codes. Our Subcommittee on Severe Accidents held meetings on these matters on March 1 and April 8, 1996. During this review, we had the benefit of discussions with representatives of the NRC staff and of the documents referenced.

<u>Conclusions and Recommendations</u>

1. Severe accident research provides information essential to the development of risk-informed regulation.

2. Severe accident research provides the basis for evaluating severe accident management strategies.

3. The NRC nuclear safety research program budget continues to decline, and various research efforts are being reduced or eliminated. Periodic analysis should be performed to assure that the remaining severe accident research efforts are focused on topics that have the greatest impact on risk and the associated uncertainties. Criteria should be developed for determining when programs have met their objectives.

4. Results of the severe accident research have shown that there is no threat of prompt containment failure posed by direct containment heating (DCH) in Westinghouse large dry containments, alpha-mode steam explosions, and Mark I liner melt-through. Research should continue to:

- determine the impact of DCH on other containment types,

- develop codes to better model the hydrogen stratification and detonation,

- determine the impact of ex-vessel steam explosions on the BWR containments,

- understand the phenomenological aspects associated with molten debris coolability,

- determine the impact of fuel coolant interaction on lower head failure, and

- determine the threats posed to steam generator tubes by the natural circulation induced by the core degradation processes.

5. Quantification of uncertainties is essential to risk-informed regulation. The NUREG-1150 effort contributed significantly to the method for quantification of uncertainties. Additional effort is needed to improve understanding and quantification of phenomenological uncertainties and their impact on Level 2 PRA results. We plan to provide more specific recommendations in this area in the future, as needed.

6. The assurance of the availability of specialized experts to advise the Commission is sometimes a tacit motivation for planning research programs. We believe that such assurance is prudent and should be explicitly recognized as a criterion in the funding of research.

Discussion

We believe it is important that the staff periodically perform top down assessments of research to assure that the work supports top level objectives, to review priorities, and to identify research efforts that have reached maturity and perhaps should be discontinued. In our view, severe accident research should have the following top-level objectives:

- support assessments of severe accident risk from operating plants,

- provide a technical basis for reviewing accident management procedures,

- support the development of risk-informed regulation, and

- provide a technical basis for evaluating advanced plant designs and operational features.

Better Level 2 PRAs are needed to reduce the uncertainties associated with the assessment of the risk to public health and safety. Severe accident research provides the bases for improving Level 2 PRAs, many of which have used unnecessarily simplistic models for severe accident behavior. Severe accident research is needed to reduce the presently large uncertainties in risk assessment results that are inimical to making sound regulatory decisions.

The processes that lead to early failure of containment are of particular importance to risk. Among such processes are DCH, fuel coolant interactions, alpha-mode steam explosions, hydrogen detonations, direct contact of core debris with containment structures, and steam generator tube ruptures. Additional assessment of DCH is needed for CE, B&W, and ice condenser containments, and for BWRs. Although it appears that large dry containments and containments with igniters can accommodate hydrogen combustion without failing, we believe that stratification and the potential for local detonation needs additional investigation.

The extent to which debris can be cooled can be pivotal in determining the likelihood of containment liner failure and long-term containment basemat melt-through. Viable criteria for coolability of molten debris either in-vessel or ex-vessel have not yet been developed.

A possible disadvantage of successful in-vessel debris cooling is the potential failure of the reactor coolant system or steam generator tubes caused by overheating from the convection of hot gases. Steam generator tube ruptures that might occur as a consequence of, or coincident with, a severe accident would provide a direct path for radionuclide release from the reactor core to the environment. The NRC and industry are addressing this issue, but we believe additional thermal hydraulic and radionuclide transport code development will be required for resolution. The present NRC codes are not capable of assessing this situation.

Currently, significant information in the severe accident area is being developed in international cooperative programs. While we fully support the bilateral agreements and the Cooperative Severe Accident Research Program (CSARP), it is important for NRC that its domestic contractors maintain capability in this area. Staff and contractors who are knowledgeable of the physics and technology of severe accident phenomena will be needed to resolve complex issues in this area, to enhance the regulatory process, and to provide technical support in the event of a real accident.

Dr. Dana A. Powers did not participate in the Committee's deliberation regarding this matter.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

References:
1. Report dated August 18, 1992, from David A. Ward, Chairman, ACRS, to Ivan Selin, Chairman, NRC, Subject: Severe Accident Research Program Plan
2. U. S. Nuclear Regulatory Commission, SECY-95-004, dated January 4, 1995, from James M. Taylor, Executive Director for Operations, NRC, for the Commissioners, Subject: Status of Implementation Plan for Closure of Severe Accident Issues, Status of the Individual Plant Examinations and Status of Severe Accident Research
3. U. S. Nuclear Regulatory Commission, NUREG/CR-6109, "The Probability of Containment Failure by Direct Containment Heating in Surry," May 1995
4. Nuclear Energy Institute, NEI 91-04, Revision 1, "Severe Accident Issue Closure Guidelines," December 1994
5. Report (undated) by F. Cheung and K. Haddad, Pennsylvania State University, Subject: Steady-State Observations and Theoretical Modeling of Critical Heat Flux Pheromena on a Downward Facing Hemispherical Surface
6. Sandia National Laboratories Letter Report, "Scaling and Design Report for Lower Head Failure Experiments," May 1995
7. Secretary-General of the OECD Report, Senior Group of Experts on Severe Accident Management (SESAM), "Severe Accident Management Implementation," October 1995
8. Secretary-General of the OECD Draft Report, "Nuclear Safety Research in OECD Countries, Areas of Agreement, Areas For Further Action, Increasing Need For Collaboration," November 1995
9. Proceedings of the Specialist Meeting On Severe Accident Management Implementation, held at Niantic, Connecticut, on June 12-14, 1995, by the Committee on the Safety of Nuclear Installations, OECD Nuclear Energy Agency

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

October 21, 1996

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

Dear Chairman Jackson:

SUBJECT: THERMAL-HYDRAULICS RESEARCH PLAN

During the 435th meeting of the Advisory Committee on Reactor Safeguards, October 9-12, 1996, we reviewed the scope and approach of the Thermal-Hydraulics Research Plan of the Office of Nuclear Regulatory Research (RES). Our Subcommittee on Thermal-Hydraulic Phenomena met on September 18-19, 1996, to review this matter. During this review, we had the benefit of discussions with representatives of the NRC staff. We also had the benefit of the documents referenced.

The overall plan developed by RES to consolidate existing computational tools into a single computer code is timely and should be implemented. We agree with its objectives of standardized programming, better physics, flexibility (modularity), computational efficiency, a graphical user interface, and thorough documentation. The RES plan to review the past 22 years of experience with codes like TRAC and RELAP, as well as the successful Code Scaling Applicability and Uncertainty evaluation methodology, should help avoid some of the problems of the past. We recommend that this review also consider the French code, CATHARE, and its uncertainty evaluation methodology.

The RES staff expects to identify the key physical processes that the new code must model. Also, RES plans to determine whether TRAC-P has an architecture that will allow it to provide flexibility with respect to insertion of new models or modules and whether it has the capability to interface with other codes like CONTAIN and SCDAP. We concur in these plans and emphasize that highest priority should be given to the development of sufficient flexibility to facilitate modifications in response to future modeling challenges.

The NRC Office of Nuclear Reactor Regulation and the Office for Analysis and Evaluation of Operational Data are primary users of thermal-hydraulic codes. They should be a part of this process from the beginning. Consequently, we recommend that a code users group be instituted early in the development program.

We concur in the RES plan to incorporate the integral effects test programs at Oregon State University, the University of Maryland, and Purdue University into the overall verification and validation program. The cooperative agreement with the French authorities to obtain analytical and experimental data developed at the Grenoble facility should also prove to be valuable for validating the code. The present RES relationship with the above three universities and the French authorities should significantly enrich the proposed thermal-hydraulics research Plan. These, along with other cooperative agreements, should be pursued, independent of the final direction of the RES Plan.

We commend the staff for the development of this Plan which holds much promise to revitalize the NRC Thermal-Hydraulics Research Program.

Additional comments by ACRS Member Ivan Catton are presented below.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

Additional Comments by Ivan Catton, ACRS Member

I agree with the views of my colleagues expressed above but would like to emphasize the need for careful planning at the outset of the RES Thermal-Hydraulics Research Plan. The research program that led to the present suite of thermal-hydraulic codes was initiated in 1974 to address the large-break loss of coolant accident. The mission was well defined and the agency met its objectives.

At the outset, it was thought that a properly designed thermal-hydraulic code would be able to model all related problems. Over the years, however, experience has shown that the codes did not meet this objective; i.e., whenever we needed solutions to a new problem that was a little different, the codes were inadequate, because they could not be readily modified to accommodate the special circumstances demanded by the new problem.

The inability of the codes to address numerous new problems emphasizes the need for a different approach. There is no single code that can model all the different physical phenomena that occur in a nuclear power plant. A broader approach is needed where different modeling schemes can be tied together to successfully address the problem at hand. Further, a skilled code user, who is also knowledgeable in the field of thermal-hydraulics, is needed to decide what is important and how to implement it in a code. A code, no matter how good, will never substitute for a capable thermal-hydraulic analyst.

Some of these problems will be heavily dependent on the use of what is commonly known as computational fluid dynamics (CFD), some on the use of the kind of modeling found in today's codes, and some will require an empirical approach. There will be some problems that may even require the use of stand-alone CFD codes. Further, the development of a single code for all users may not be a realistic goal. A skilled user needs a different level of computational power than does a less-skilled user. Ensuring adequate flexibility in a single code to accommodate the needs of both computational power and user skills will require a great deal of thoughtful planning; this planning should take place at the beginning of the development of the RES Thermal-Hydraulics Research Plan.

References:

1. Memorandum dated September 6, 1996, to the Commission from James M. Taylor, NRC Executive Director for Operations, Subject: Thermal-Hydraulic Five-Year Research Plan, (Predecisional - For Internal Use Only)
2. U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Analytical Support Group, Technical Analysis Reports, SASG-94-01 - SASG-94-05; SASG-95-01 - SASG-95-07; SASG-96-01 - SASG-96-07 (Proprietary Information)
3. ACRS report dated June 15, 1989, from David A. Ward, ACRS Chairman, to Lando W. Zech, Jr., NRC Chairman, Subject: NRC Thermal-Hydraulic Research Program
4. ACRS report dated June 7, 1988, from David A. Ward, ACRS Chairman, to Lando W. Zech, Jr., NRC Chairman, Subject: NRC Research Related to Heat Transfer and Fluid Transport in Nuclear Power Plants

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

November 19, 1996

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

Dear Chairman Jackson:

SUBJECT: POSITION ON DIRECTION SETTING ISSUE 22 -- FUTURE ROLE OF NRC RESEARCH

During the 435th and 436th meetings of the Advisory Committee on Reactor Safeguards, October 9-12 and November 7-9, 1996, respectively, we reviewed Direction Setting Issue (DSI) 22. At the 435th meeting, we discussed this issue with the NRC staff. We also had the benefit of the documents referenced.

Direction Setting Issue 22 raises the question of what role the Office of Nuclear Regulatory Research (RES) will have in the future. A range of possible roles is defined in the discussion. These vary from elimination of a research capability at NRC to continuation of the research at its current, diminished level on a broad range of topics. The preliminary thinking is to select the continuing "business as usual" role for RES.

We contend that, first, changes are occurring within both the nuclear industry and the regulatory community that make it essential for NRC to have a research function. Second, we contend that a "business as usual" approach to NRC research is too timid. There is an urgency for the NRC to have research information to meet its obligations to protect the public health and safety in a changing environment. Finally, we contend that the planning for future research should be directed toward areas of focused need. In particular, research is needed to support NRC's transition to risk-informed and performance-based regulation.

The research arm of NRC has occupied a central role in the development of the body of regulations needed to ensure public health and safety in the commercial use of nuclear power. Since the division of the Atomic Energy Commission into the NRC and what eventually became the Department of Energy, RES has overseen the work needed to develop the design-basis analysis of nuclear power plants. This has included ensuring through a combination of experimental and analytical research that the analyses done for

Appendix K to 10 CFR Part 50 are on a sound technical foundation. RES has also undertaken a vast effort to understand the residual risk posed by the use of nuclear power through the studies of severe accidents and the associated radionuclide source terms. RES has, in fact, been responsible for the evolution in the analysis of reactor safety from the bounding and the qualitative to the use of quantitative risk analysis.

From the pinnacle following the accident at Three Mile Island, RES has suffered a continuing scale-back of the activities it can afford to undertake. As with many institutions facing budgetary pressures, the longer term benefits of research activities have been sacrificed to ensure that there is the necessary financial backing for day-to-day activities that are the responsibility of NRC. NRC's research budget has, then, suffered disproportionately when funding cutbacks have been inflicted on NRC as a whole. Today, the available funding for research is, indeed, small enough that it is a legitimate question whether a viable research program can be maintained.

At the time these cutbacks in research funding have been taking place, changes have also been taking place in the way society deals with safety regulation. Most directly obvious has been the effort supported by both the Executive Branch and by Congress to base regulation on actual risk rather than bounding conservatism. The Vice President heads a Government-wide effort to base regulation, including regulation of nuclear power, on risk. Relative to most other regulatory agencies, NRC is well on the way to developing a risk-informed and performance-based regulatory system. NRC may well set an example for other regulatory agencies in this regard. It is, then, important that this be a good example.

A second societal development that will have safety implications is the economic deregulation of electrical power generation. This development has yet to be fully realized, but already efforts are being undertaken by the nuclear utilities to achieve greater economic competitiveness. Increases in reactor operating power and the extension of fuel life are just two immediate steps the industry is taking that have obvious safety implications. It is widely forecast that draconian measures will be necessary in the future to maintain nuclear power as a viable option for the generation of electrical energy. There are, of course, other changes taking place in the industry that fall in the domain of NRC such as plant aging; plant decommissioning; development of new, passive plant designs; and disposal of nuclear waste.

NRC is making great efforts to respond to the challenges posed by societal and industrial changes that are now taking place. The information available to the agency to meet these challenges is, however, proving to be limiting. By way of examples, consider the following:

- NRC is attempting to develop a risk-informed and performance-based regulatory system to improve the safety of nuclear plants and to relieve the industry of unnecessary burden. But, NRC is trying to do this without any detailed knowledge of shutdown risk because RES is unable to fund studies comparable to the NUREG-1150 studies performed to understand risk during power operations.

- NRC development of probabilistic methods has not kept pace with its needs. Methods to treat human errors of commission or the impacts of organizational factors and management practices on risk are not available. Experience shows that human errors, organization, and management are responsible for or contributing to many accidents and "near misses."

- NRC wants to regulate in light of risk, but there is now only the technical capability for routine, noncontroversial evaluation of core-damage frequency. The capacity to extend estimates of core-damage frequency to evaluate risk has not been made widely available. There is not even consensus on how accurately analyses of risk, given that core damage has occurred, must be done nor how comprehensive such analyses must be.

- The introduction of digital instrumentation and control (I&C) systems in nuclear power plant safety related systems requires NRC to have the capability to regulate high-reliability software-based systems. NRC's understanding is limited to current software engineering methods which employ highly disciplined development process to design and produce high-reliability software. A consequence of this approach is lack of well-developed methods for evaluating the product of the process. NRC is limited to regulating the process of design and development of digital I&C systems because no accepted tools are available for evaluating the product.

- Financial constraints forced NRC to allow its codes for predicting fuel behavior to atrophy so they are no longer up to the state of the art. These codes cannot adequately predict fuel and clad behavior at burnups now being used by licensees. Recovery actions by RES have been constrained by resource limitations to narrow topical areas.

- NRC has not yet been able to formulate a risk-informed and performance-based fire protection rule to replace Appendix R to 10 CFR Part 50 which has been the source of so many exemptions and other controversies.

- NRC's opportunities to leverage its research budget by participating in international research consortia are becoming

limited as NRC has less to contribute to the consortia efforts.

- NRC finds it must evaluate new, passive plant designs using tools developed for older plant designs because it cannot afford to develop analytical tools better suited for the simulation of the physics of these new designs. NRC must "make do" with computational tools that are now over a decade old and don't even begin to take advantage of all the more recent advances in computer technology.

- NRC is unable to predict or detect newly discovered modes of degradation of the primary pressure boundaries of pressurized water reactors. It has been forced to use rules designed to deal with wastage and corrosion of steam generator tubes to protect against a variety of forms of stress corrosion cracking.

There is clearly a need for a more aggressive NRC research program to confront the many challenges that the agency continues to face. We can be confident that the agency will meet its obligation to protect the public health and safety. But, without up-to-date tools produced by a forward thinking research organization, the agency will have to resort to methods that do not contribute to either regulatory efficiency or economic efficiency of the nuclear industry.

The financial resources now available to the agency for performing research are indeed limited. It has been necessary to make hard choices on what is to be done and what must be abandoned. A significant factor in the thinking on what is to be supported and what is to be abandoned has been a desire to preserve technical capability. This effort to preserve technical capability appears to have:

- led to an emphasis on the things that the agency knows best such as the thermal hydraulics of existing reactors, and

- diluted the efforts in many areas to preserve the current organizational units of RES.

The preliminary decision for DSI 22, which is to continue conducting research as it has been done in recent years, appears to enforce this emphasis on what is known well and to preserve the existing organizational structure of RES.

It is our position that more aggressive options need to be developed in response to DSI 22. One of these options is to focus the research in areas to meet the agency needs as it embarks on its experiment with risk-informed and performance-based regulation. The goal of research, then, ought to be, first, to provide risk

information that is far more comprehensive than that now available, and then, to identify the performance indicators that do indeed reflect the risk. Furthermore, efforts are needed to use plant data and event reports to assess the adequacy of current probabilistic risk assessment methods.

RES also needs to anticipate safety implications that licensees will make in response to economic pressures. RES should be in a position to provide tools suitable for the safety evaluation of these changes. To do this, the split of work by RES between user requests and self-directed work may have to be reevaluated.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

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1. United States Nuclear Regulatory Commission, "Strategic Assessment and Rebaselining Initiative, Stakeholder Involvement Process Paper," dated September 16, 1996
2. United States Nuclear Regulatory Commission, "Strategic Planning Framework," dated September 16, 1996
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

October 22, 1996

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

Dear Chairman Jackson:

SUBJECT: CAPABILITY OF THE NRC SCDAP/RELAP5 CODE TO PREDICT TEMPERATURES AND FLOWS IN STEAM GENERATORS UNDER SEVERE-ACCIDENT CONDITIONS

During the 434th and 435th meetings of the Advisory Committee on Reactor Safeguards, September 12-13 and October 9-12, 1996, respectively, we held discussions with representatives of the NRC staff concerning the capability of the SCDAP/RELAP5 code to predict steam generator tube temperatures and flows under certain severe-accident conditions. An ACRS member attended a meeting on August 19-20, 1996, of the NRC-sponsored experts panel, which reviewed the adequacy of SCDAP/RELAP5 for the above conditions. We also had the benefit of the documents referenced.

Under some severe-accident conditions, natural convection carries hot steam and gases from the core through the hot leg and into the steam generator inlet plenum. Some fraction of the flow then goes from the inlet plenum through some of the steam generator tubes to the exit plenum and returns via the remaining tubes to the inlet plenum where it mixes with the flow from the hot leg. Countercurrent stratified flow occurs in portions of the core, in the hot leg, and in the steam generator inlet plenum. Either the hot-leg piping, the inlet-surge line, or the steam generator tubes are projected to eventually fail by high-temperature creep rupture. A failure of any one of these components will lead to depressurization of the reactor and probably preclude additional failures. The risk significance of such failure depends on which component fails first. If the steam generator tubes fail first, a containment bypass path could be created for radionuclide release directly to the environment. Such a scenario could be a significant contributor to risk.

In support of the steam generator integrity rulemaking, the NRC staff is using the SCDAP/RELAP5 code to examine steam generator tube integrity for severe-accident scenarios. Steam generator tube temperatures under these conditions are strongly dependent on the extent of mixing of the hot fluid entering the inlet plenum of the steam generator with the cold return fluid and the fraction of steam generator tubes that carry the hot fluid to the exit plenum of the steam generator. These phenomena cannot be

predicted mechanistically by a one-dimensional (1-D) lumped parameter code such as SCDAP/RELAP5, because they depend on the details of the countercurrent flow in the hot leg, the hot plume flow pattern in the steam generator inlet plenum, and the characteristics of the entire recirculating flow.

If a 1-D lumped parameter code is to be used to analyze the above conditions, the key phenomena must be determined either by suitable supporting analyses or from experiments and then provided as input to the code. The NRC staff and its contractor have used the results of the 1/7-scale tests conducted by Westinghouse to "tune" the SCDAP/RELAP5 code and they have demonstrated that the code can adequately reproduce a limited subset of the test results. These 1/7-scale tests appear to be reasonably well designed and conducted. A panel of experts was convened by the NRC staff to review the adequacy of SCDAP/RELAP5 and the scaling analyses of the tests. Although the panel raised some questions that have not been addressed, it determined that SCDAP/RELAP5 is appropriate for predicting steam generator tube temperatures under severe-accident conditions.

However, we did not find the scaling analyses of the 1/7-scale tests to be completely satisfactory. The tests lack geometric similitude, i.e., the steam generator tubes are a factor of three too large and there are too few tubes. Additionally, the appropriateness of the dimensionless parameters used to scale the tests is questionable. Furthermore, fully developed forced-flow heat transfer correlations were used to represent conditions of mixed convection and developing forced flow, and radiative heat transfer was neglected.

The staff noted that the timing of tube failure is very sensitive to the tube temperatures. Such sensitivity suggests that the uncertainties in the temperature calculations need to be explicitly identified and their impact on this timing assessed. We believe that present NRC codes can be used for assessing uncertainties in the timing of component failures, if proper judgment is exercised by analysts to evaluate code results. We recommend that an appropriate uncertainty analysis addressing the above concerns, including the effects of radionuclide transport, be performed.

In our June 28, 1996 report, we stated that the present NRC codes were not capable of assessing steam generator tube ruptures under severe-accident conditions. Having had the opportunity to review the reports of the panel members and having had more detailed presentations on the use of the 1/7-scale tests to "tune" the SCDAP/RELAP5 code, we now believe that it can be used for the analyses required to support the development of the steam generator integrity rule.

We commend the staff for its competent and timely response to our earlier concerns and look forward to additional interactions on this important topic.

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. Report dated June 28, 1996, from T. S. Kress, ACRS Chairman, to Shirley Ann Jackson, NRC Chairman, Subject: Severe Accident Research
2. Memorandum dated September 7, 1996, from I. Catton, ACRS Member, to ACRS Members, Subject: Conditional Probability of a Steam Generator Tube Rupture Following a Core Damage Accident
3. Memorandum dated August 27, 1996, from P. Griffith, Member of NRC-Sponsored Experts Panel, Massachusetts Institute of Technology, to Khatib-Rahbar, Energy Research, Inc., regarding Capability of NRC SCDAP/RELAP5 Code
4. Memorandum dated August 30, 1996, from M. Ishii, Member of NRC-Sponsored Experts Panel, Purdue University, to Richard Lee, Office of Nuclear Regulatory Research, NRC, regarding Capability of NRC SCDAP/RELAP5 Code
5. Letter dated September 11, 1996, from R. Viskanta, Member of NRC-Sponsored Experts Panel, Purdue University, to Khatib-Rahbar, Energy Research, Inc., Subject: SCDAP/RELAP5 Code Modeling of Natural Circulation Under Severe Accident Conditions, Fauske & Associates, Inc., Burr Ridge, Illinois, August 19-20, 1996

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

November 18, 1996

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

Dear Chairman Jackson:

SUBJECT: PLANT-SPECIFIC APPLICATION OF SAFETY GOALS

During the 436th meeting of the Advisory Committee on Reactor Safeguards, November 7-9, 1996, we discussed the application of Safety Goals on a plant-specific basis. This subject was also discussed at meetings of our Joint Subcommittees on Probabilistic Risk Assessment and Plant Operations on July 17-18, 1996, and of our Subcommittee on Probabilistic Risk Assessment on August 7, 1996. We also had the benefit of the documents referenced.

In a Staff Requirements Memorandum dated June 11, 1996, we were requested to provide recommendations on how the Commission's Safety Goals and Safety Goal Policy should be revised to make them acceptable for use on a plant-specific basis.

The Safety Goal Policy Statement made it clear that the Quantitative Health Objectives (QHOs) and the subsidiary Core Damage Frequency (CDF) goal were to provide standards for the NRC staff to judge the overall effectiveness of the regulatory system. That is, if the risk posed by the population of plants on the average proved to be less than the Safety Goals, then the staff (and presumably the public) would deem that the regulatory system had functioned appropriately to protect the health and safety of the public.

The Safety Goals quantified "how safe is safe enough" for the population of U. S. plants. For an individual plant, however, the acceptable level of risk is determined by the concept of "adequate protection," which in the final analysis means compliance with the body of regulations. Risk-informed analyses would provide a more rational basis for making regulatory decisions regarding plant-specific requests for exemptions from the rules or for changes to the licensing basis, and the acceptability of new regulations.

In our August 15, 1996 report, we stated: "the safety goals and subsidiary objectives can and should be used to derive guidelines for plant-specific applications. It is, however, impractical to rely exclusively on the Quantitative Health Objectives (QHOs) for routine use on an individual plant basis. Criteria based on core

damage frequency (CDF) and large, early release frequency (LERF) focus more sharply on safety issues and can provide assurance that the QHOs are met."

In developing plant-specific criteria, it is important to consider the regulatory needs in the near future and to ensure that the process will be evolutionary rather than so revolutionary that it might discourage the licensees from using this approach. It appears that most of the anticipated licensee requests for changes to their current licensing basis will deal with Level 1 probabilistic risk assessment (PRA) issues, e.g., inservice inspection, extension of allowed outage times. Furthermore, most licensees have only recently familiarized themselves with Level 1 PRA methodology for the narrow regime of power operations. They are just beginning to integrate findings of such Level 1 risk assessments with the safe operation of their plants. Even the NRC staff is still coming to grips with the implications of Level 1 risk assessment results for regulation of nuclear plants. Many licensees do not have access to the technologies for facile conduct of full-scope Level 2 or Level 3 PRAs that treat power operations, low power/shutdown operations, as well as accidents initiated by external events. Commonly accepted standards for such extensive, in-depth analyses do not exist.

An evolutionary and pragmatic approach for using Safety Goals on a plant-specific basis would be to use the CDF as the primary criterion for evaluating proposed changes along with a qualitative or quantitative evaluation of the possible Level 2 and Level 3 PRA issues raised by these changes. For a quantitative analysis, the following two options are offered:

1) Full-scope Level 2 PRA (with fission product transport capability).

To use this option, a conservative value for a LERF criterion must be determined. This value, along with the CDF criterion, will provide an acceptable basis for decisionmaking. We note that both the NRC staff and the Electric Power Research Institute, in its, "PSA Application Guide," are proposing the use of LERF as an acceptance criterion.

2) Full-scope Level 2 PRA (without fission product transport capability).

To use this option, conservative values for early containment failure frequency criteria for different reactor designs must be determined. These values, along with the CDF criterion, will provide an acceptable basis for decisionmaking.

In the longer term, we believe the agency should move beyond the evaluation of risk associated with proposed changes to individual plant licenses and apply the Safety Goals to assess the

acceptability of plant-specific risk. This could be done in terms of the QHOs, along with the CDF, or in terms of the CDF and LERF. To use the QHOs directly, it would be necessary to have full-scope Level 3 PRAs. We believe that the use of Level 3 PRAs in the future should be encouraged.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

References:

1. Staff Requirements Memorandum dated June 11, 1996, from John Hoyle, Secretary, NRC, to John T. Larkins, Executive Director, ACRS, Subject: Meeting with ACRS, Friday, May 24, 1996
2. ACRS report dated August 15, 1996, from T. S. Kress, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: Risk-Informed, Performance-Based Regulation and Related Matters
3. Electric Power Research Institute Report TR-105396, "PSA Application Guide," prepared by ERIN Engineering and Research, Inc., August 1995

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This compilation contains 47 ACRS reports submitted to the Commission, or to the Executive Director for Operations, during calendar year 1996. 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, the U. S. Library of Congress, and the Internet at http://www.nrc.gov/ACRSACNW. 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.

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