NUREG-1125
Volume 9

A Compilation of Reports of **The Advisory Committee on Reactor Safeguards**

1987 Annual

U.S. Nuclear Regulatory Commission

April 1988

NOTICE

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Most documents cited in NRC publications will be available from one of the following sources:

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Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at the NRC Library, 7920 Norfolk Avenue, Bethesda, Maryland, and are available there for reference use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from the American National Standards Institute, 1430 Broadway, New York, NY 10018.

ABSTRACT

This compilation contains 47 ACRS reports submitted to the Commission or to the Executive Director for Operations during calendar year 1987. It also includes a report to the Congress on the NRC Safety Research Program for FY 1988. All reports have been made available to the public through the NRC Public Document Room and the U.S. Library of Congress. The reports are divided into two groups: Part 1: ACRS Reports on Project Reviews, and Part 2: ACRS Reports on Generic Subjects. Part 1 contains ACRS reports alphabetized by project name and within project name by chronological order. Part 2 categorizes the reports by the most appropriate generic subject area and within subject area by chronological order.

PREFACE

The enclosed reports represent the recommendations and comments of the U.S. Nuclear Regulatory Commission Advisory Committee on Reactor Safeguards during calendar year 1987. This publication, Volume 9, is an annual supplement to NUREG-1125. Previous issues of NUREG-1125 are as follows:

Volume	Inclusive Dates
1 through 6	September 1957 through December 1984
7	Calendar Year 1985
8	Calendar Year 1986

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Part 1: ACRS Reports on Project Reviews

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

March 9, 1987

The Honorable Edward J. Markey
Committee on Energy and Commerce
United States House of Representatives
Washington, D.C. 20515

Dear Congressman Markey:

We note your interest in our ongoing deliberations relative to the Seabrook Station, as evidenced by your letter of February 26, 1987 to Mr. David A. Ward, ACRS.

Section 182b of the Atomic Energy Act requires the Advisory Committee on Reactor Safeguards to "review each application ... for ... an operating license for a facility...." The Committee issued a report, dated April 19, 1983, with respect to the proposal to operate the Seabrook Station; a copy of that report is attached. In the report, we indicated that there were some open issues, and we noted the absence of a fully developed emergency plan. Because the Committee reported a satisfactory conclusion only with respect to operation of the plant at power levels at or below 5 percent of full power, our review of the Seabrook Station operating license is not complete.

We have begun a review of matters associated with emergency planning for the Seabrook Station. When we have completed our work, and fulfilled our obligation to provide sound and dispassionate advice to the Commission, that advice will be publicly available, as will the listing of inputs that contributed to it. We will provide you with a copy of our report at that time.

Sincerely,

W Kerr

William Kerr
Chairman

Attachment:
Letter from J.C. Ebersole, Acting Chairman, ACRS, to N.J. Palladino, Chairman, NRC, dated April 19, 1983

cc: Honorable Philip R. Sharp, Chairman
Subcommittee on Energy and Power

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

April 19, 1983

Honorable Nunzio J. Palladino
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Dr. Palladino:

SUBJECT: ACRS REPORT ON LOW POWER OPERATION OF THE SEABROOK STATION, UNITS 1 AND 2

During its 276th meeting, April 14-16, 1983, the Advisory Committee on Reactor Safeguards reviewed the application of the Public Service Company of New Hampshire, acting as agent for and on behalf of the Seabrook Owners Group (the Applicant), for an operating license for the Seabrook Station, Units 1 and 2. The station is to be operated by the Public Service Company of New Hampshire. This application was considered at an ACRS Subcommittee meeting in Hampton Beach, New Hampshire, on April 1-2, 1983. Members of the Subcommittee toured the facility on April 1, 1983. In our review, we had the benefit of discussions with representatives of the Applicant, the Yankee Atomic Electric Company, Westinghouse Electric Corporation, United Engineers and Constructors, Inc., the NRC Staff, and with members of the public. We also had the benefit of the documents listed below. The Committee commented on the construction permit application for Seabrook Station, Units 1 and 2 in a report dated December 10, 1974.

The Seabrook Station is located on the western side of Hampton Harbor, in the Township of Seabrook, Rockingham County, New Hampshire, approximately 11 miles south of Portsmouth, New Hampshire and 40 miles north of Boston, Massachusetts.

Each Seabrook unit uses a Westinghouse nuclear steam supply system with a rated core power of 3411 MWt. The containment for each unit consists of a steel lined, reinforced concrete structure which is surrounded by a reinforced concrete containment enclosure. The design pressure of the containment is 52 psig. The annular space between containment and enclosure is maintained at a slight negative pressure.

Seabrook will use Westinghouse Model F steam generators, which incorporate design changes intended to eliminate the problems experienced with earlier models. We wish to be kept informed concerning the performance of these steam generators.

We were favorably impressed by the amount of attention given and resources expended in the area of personnel training. The result appears to be an excellent educational system for operations personnel, including operators and technicians. The resources at the disposal of the Applicant, including those of the Yankee Atomic Electric Company, appear to be appropriate for the operation of this nuclear power station.

The ACRS has on several occasions recommended that evaluations be made of the capability of light water nuclear power plants to be shut down safely in the event of an earthquake of greater severity and lower likelihood than the safe shutdown earthquake. The implications of recent seismic activity, such as the January 1982 earthquakes in central New Brunswick and New Hampshire, are being evaluated. We recommend for the Seabrook Station that specific attention be given to the seismic capability of those components that are important to the accomplishment of safe shutdown including the emergency AC power supplies, the DC power supplies, and small components such as actuators and instrument lines.

The Applicant has undertaken a full-scope probabilistic risk assessment (PRA) which is scheduled for completion about October 1983. The ACRS wishes to be kept informed concerning the results of the NRC Staff's review and evaluation of this PRA.

The Seabrook Station, Units 1 and 2 will be the first commercial nuclear power plant in the state of New Hampshire; the Station is also situated very close to the New Hampshire-Massachusetts border. As a result, the NRC Staff and Applicant must give particular attention to assuring proper coordination with appropriate state and regional agencies in the development of effective emergency plans. There is a large summertime increase in population within a few miles of the site due to the beach areas of Seabrook and Hampton, New Hampshire. The nature of the road network serving the beach requires that special attention be given to the problems associated with evacuation. Because the emergency plan is not yet fully developed, we were unable to review it.

A number of other items have been identified by the NRC Staff as Outstanding Issues. There is also a set of Confirmatory Issues that awaits additional documentation. We found no reason to believe that any of these issues will be especially difficult to resolve. We recommend that they be resolved in a manner satisfactory to the NRC Staff.

Fuel loading for Unit 1 is scheduled for September 1984 and fuel loading for Unit 2 is planned to take place about 2.5 years after fuel loading for Unit 1. Should there be a significant delay in this schedule, we would expect to examine the need for additional review of Unit 2.

We believe that, if due regard is given to the items mentioned above, and subject to satisfactory completion of construction, staffing, and preoperational testing, there is reasonable assurance that the Seabrook

Station, Units 1 and 2, can be operated at core power levels up to 5 percent of full power without undue risk to the health and safety of the public.

Sincerely,

Jesse C. Ebersole

Jesse C. Ebersole
Acting Chairman

References:

1. Public Service Company of New Hampshire, Seabrook Station "Final Safety Analysis Report," Volumes 1-15, with Amendments 45-48
2. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Operation of Seabrook Station, Units 1 and 2," NUREG-0896, dated March 1983.
3. Written Public Comments from J. Doughty, Seacoast Anti-Pollution League (SAPL), Subject: SAPL Comments to the Advisory Committee on Reactor Safeguards Subcommittee Conducting the Independent Technical Review for the Seabrook Nuclear Power Plant, April 1983, received April 1, 1983.
4. Written Public Comments from Rep. Roberta C. Pevear, New Hampshire House of Representatives, Subject: Statement Before Advisory Committee on Reactor Safeguards Meeting on Seabrook Operating License, April 2, 1983, received April 2, 1983.
5. Written Public Comments from Elizabeth Dolly Weinhold, Subject: Seismic Issues, received April 2, 1983.
6. Written Public Comments from Rep. Roberta C. Pevear, New Hampshire House of Representatives, Subject: Response to Kulash Report on evacuation planning, dated April 4, 1983.
7. Written Public Comments from Diana P. Sidebotham, President, New England Coalition on Nuclear Pollution, Inc., Subject: Remarks Prepared for delivery at April 1, 1983 Subcommittee meeting on Seabrook Station, dated April 11, 1983.

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

August 11, 1987

The Honorable Lando W. Zech, Jr.
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Zech:

SUBJECT: ACRS REVIEW OF APPLICATION FOR PRELIMINARY APPROVAL OF THE WESTINGHOUSE RESAR/SP-90 DESIGN

During the 328th meeting of the ACRS, August 6-8, 1987, we heard a presentation by the NRC Staff describing its schedule for review of the subject application. We regard this program as highly important and expect to participate extensively in the review. We believe it will be appropriate for the ACRS subcommittee on Westinghouse reactor plants to meet with the NRC Staff and the Westinghouse representatives in a series of meetings beginning well in advance of the date when a draft SER is expected to be available.

Among the subjects that we would like to review are the scope and results of the probabilistic risk assessment used by Westinghouse in support of its design; a comparison of the design with modern plants of similar type, both in the U.S. and abroad, for example, the Vogtle, Paluel, Sizewell B, and KONVOI plants and the Japanese APWR; and similar comparison with the EPRI requirements document for future PWRs to the extent feasible. We will be interested in the reasons for design choices and the NRC Staff evaluations of these choices.

Sincerely,

W Kerr

William Kerr
Chairman

Part 2: ACRS Reports on Generic Subjects

2

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

June 9, 1987

The Honorable Lando W. Zech, Jr.
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Zech:

SUBJECT: ACRS COMMENTS ON DRAFT NUREG-1226, "DEVELOPMENT AND UTILIZATION OF THE NRC POLICY STATEMENT ON THE REGULATION OF ADVANCED NUCLEAR POWER PLANTS"

During the 326th meeting of the ACRS, June 4-6, 1987, and in our 325th meeting, May 7-9, 1987, we discussed NUREG-1226, "Development and Utilization of the NRC Policy Statement on the Regulation of Advanced Nuclear Power Plants." A Subcommittee meeting was also held to discuss this NUREG with the NRC Staff on April 24, 1987. During our discussion, we had the benefit of the documents referenced and also of earlier meetings with the NRC Staff. We had previously reviewed the Advanced Reactor Policy Statement and had commented on the statement in a letter to Chairman Palladino dated October 16, 1985.

When the Advanced Reactor Policy Statement was issued, in July 1986, the Commission directed the NRC Staff to prepare a document that would describe its development. Later the purpose of the document (which became NUREG-1226) was extended to include factors important to implementation of the policy. Our comments will be limited to the implementation aspects of the document. We are in general agreement with the implementation approach, but have several comments.

The early interactions between the Staff and an applicant are to be concerned with review of conceptual design, well in advance of any formal application for a construction permit or a design certification. The Staff reported that it intends to assure a conceptual design that looks ahead to possible future standardization. We concur.

The implementation plan encourages, but does not require, the development of new designs based on building and operation of prototypes. We believe that operation of prototypes prior to certification of designs should be the norm and the only exceptions should be made in carefully evaluated cases, where there exists a sufficiently well-developed experience base.

NUREG-1226 uses the terms "defense-in-depth" and "design-basis accident." These are time-honored terms, but they are inexact as concepts. For example, there is a requirement to consider "beyond design basis" scenarios in the design. This presents, at minimum, a serious semantic problem. We believe the Staff needs to clarify its use of these terms.

The policy statement encourages use of "performance-based" rather than "prescriptive" requirements. Again we have concerns that these terms are used without being well defined. For example, 10 CFR 50.46 is certainly a performance-based requirement for the design of an Emergency Core Cooling System (ECCS), but prescriptions for analyzing performance are given in excruciating detail in Appendix K. We believe there is a need to clarify both of these terms and concepts.

We believe the attribute "simplicity" is not always a virtue to be encouraged in future nuclear power plants. From the perspective of safety it is important to have plant systems designed to be easy to operate, easy to maintain, easy to understand, and capable of accommodating a broad spectrum of challenges. However, simplicity does not always provide these characteristics. As an example, increased automation, as a means to make a plant easier to operate, may actually make the design more complex. The history of the evolution of engineered systems indicates they often become more complex as they are improved in reliability and performance, including safety performance.

We believe that NUREG-1226 should provide more definitive guidance for sabotage-protection considerations for advanced plant designs. We recognize this as a difficult issue, and it is for this reason that the Staff should give it additional attention.

Additional remarks by ACRS Member David Okrent are presented below.

Sincerely,

W Kerr

William Kerr
Chairman

Additional Remarks by ACRS Member David Okrent

I believe that defense-in-depth should be maintained such that an appropriate containment or other system intended to mitigate severe core melt accidents will be provided.

References:
1. U.S. Nuclear Regulatory Commission, NUREG-1226, "Development and Utilization of the NRC Policy Statement on the Regulation of Advanced Nuclear Power Plants," draft published May 5, 1987.
2. U.S. Nuclear Regulatory Commission, SECY-85-279, Subject: "Revised Advanced Reactor Policy Statement," dated August 21, 1985.
3. U.S. Nuclear Regulatory Commission, "Regulation of Advanced Nuclear Power Plants, Statement of Policy," 51 FR 24643, dated July 8, 1986.

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

October 15, 1987

Mr. Victor Stello, Jr.
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Stello:

SUBJECT: ACRS COMMENTS ON NUCLEAR POWER PLANT AIR COOLING SYSTEMS

During the 330th meeting of the ACRS, October 8-10, 1987, we discussed a report from our Subcommittee on Auxiliary Systems regarding heating, ventilating, and air conditioning system failures and their impact on safety systems. This matter was discussed on June 27, 1986 during a joint meeting of the ACRS Subcommittees on Occupational and Environmental Protection Systems and on Auxiliary Systems. It was also discussed by the Auxiliary Systems Subcommittee during a meeting held on October 1, 1987. The Subcommittees had the benefit of discussions with representatives of the Office of Nuclear Reactor Regulation and the document referenced.

During the June 27, 1986 meeting, representatives of the NRC Staff stated that failures of air cooling systems for areas housing key components (for example, RHR pumps, switch gear, diesel generators, etc.) in certain nuclear power plants contribute significantly to estimated core-melt frequencies.

Because corrective measures are often taken once potential cooling system failures are identified, the impact of these potential failures on the proper functioning of these systems has not been reflected in the final PRAs issued for these plants. As a result, some members of the NRC Staff and some licensees whose plants have similar deficiencies may not be aware of these problems.

Based on these observations, we recommend that the NRC Staff examine the extent to which these problems may be generic and take any corrective actions deemed necessary.

Sincerely,

W Kerr

William Kerr
Chairman

Reference:
Presentation material provided by Arthur Buslik, Office of Nuclear Reactor Regulation, before a joint meeting on June 27, 1986 of the ACRS Subcommittees on Occupational and Environmental Protection Systems and on Auxiliary Systems.

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

January 14, 1987

Mr. Victor Stello, Jr.
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Stello:

SUBJECT: ACRS COMMENTS ON THE DRAFT NRC REPORT ON THE IMPLICATIONS OF THE ACCIDENT AT CHERNOBYL NUCLEAR STATION UNIT 4

During the 321st meeting of the ACRS, January 8-10, 1987, we considered the implications of the accident at the Chernobyl nuclear station Unit 4 as it relates to nuclear power plants in the United States. This subject was also considered during our 320th meeting, December 11-13, 1986 and our 319th meeting, November 6-8, 1986. In our review, we had the benefit of meetings of our Subcommittee on Safety Philosophy, Technology, and Criteria held on November 5 and December 10, 1986, of discussions with the NRC Staff, and of the documents referenced.

We have seen preliminary drafts of the NRC Staff's report on lessons learned from the Chernobyl Nuclear Station accident. We agree that a thorough evaluation of the accident should be performed and that the lessons learned should be applied in the NRC regulatory process.

Although we have not reviewed the Staff's report in detail (nor have we seen a final draft), we consider the proposals made in the drafts we have seen to be sound and have no suggestion for any major changes in direction.

Sincerely,

W Kerr

William Kerr
Chairman

References:

1. Memorandum dated December 5, 1986 from Themis P. Speis, NRC, to Raymond Fraley, ACRS, Subject: Chernobyl Information for ACRS Review, with enclosure: revised draft of the Chernobyl Implications Assessment Report dated December 4, 1986
2. Memorandum dated October 30, 1986 from Themis P. Speis, NRC, to Raymond Fraley, ACRS, Subject: Chernobyl Information for Subcommittee Review, with enclosure: Draft Chernobyl Implications Assessment: Assessments of Candidate Issues dated October 30, 1986

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

January 15, 1987

The Honorable Lando W. Zech, Jr.
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Chairman Zech:

SUBJECT: ACRS COMMENTS ON THE IMPLICATIONS OF THE ACCIDENT AT THE CHERNOBYL NUCLEAR STATION UNIT 4

During the 321st meeting of the ACRS, January 8-10, 1987, we considered the implications of the accident at the Chernobyl nuclear station as it relates to nuclear power plants in the United States. This subject was also considered during our 320th meeting, December 11-13, 1986 and our 319th meeting, November 6-8, 1986. In our review, we also had the benefit of meetings of our Subcommittee on Safety Philosophy, Technology, and Criteria held on November 5 and December 10, 1986, and discussions with the NRC Staff.

The Chernobyl accident reminds us that, although a large nuclear power plant accident somewhere in the United States is unlikely, it is not impossible. We believe it is essential that a thorough evaluation of the Chernobyl accident be performed and any important lessons from this evaluation are used in evaluating the risk posed by domestic nuclear power plants. We recognize that the NRC Staff has such a program under way.

We believe that the most important lesson to be learned from the Chernobyl accident is that high priority must be given to ensuring that the management and the operating staff of each plant are competent and are motivated to operate the plant safely and in strict compliance with plant administrative controls. Strong emphasis should be given to the adequacy of the training and to the ability of the responsible personnel to prevent, to manage, and to mitigate severe accidents. The operating staff should include on-site personnel with engineering capability who fully understand the design and operating characteristics of the plant and the implications for plant safety. Such a staff should know the basis for the engineering and safety decisions made during plant design. Although these recommendations are not new, the Chernobyl accident has reemphasized their importance.

Chernobyl also reinforces the known importance of determining the extent to which containments are capable of dealing with accidents more severe than the currently specified "design basis accidents." We recommend that the NRC Staff give continued high priority to its current effort to examine the containment performance expected for operating nuclear power plants and to examine improvements needed to ensure that risk is limited to an appropriate level.

Reactivity transients severe enough to damage a light-water-reactor core can be hypothesized. Risk estimates, operating experience, and informed opinion all indicate that such transients are very unlikely. However, such estimates and opinions depend in part upon assumptions that personnel will comply with the administrative controls for operation, rather than depending entirely upon inherent characteristics of the hardware and processes. Present methods of risk assessment do not satisfactorily account for personnel errors of the sort that could lead to noncompliance with such administrative controls. Operating experience cannot be extensive enough to give high assurance that such errors are incredible. For these reasons, there should be a systematic reexamination of the potential for severe reactivity transients, with emphasis on the impact of human error. Multiple rod ejection, cold water insertions, void collapse, boron depletion, inappropriate bypassing of exposed safety circuits, and the importance of positive temperature coefficients during early core life are examples of the events and conditions that should be restudied. The levels of defense against severe reactivity transients should be identified and, if possible, appropriately codified.

Emergency response following the Chernobyl accident confirmed the need to ensure that the Protective Action Guides developed for application in the United States are comparable with those in neighboring countries and the need to reexamine the national policy on the storage and use of radioprophylactic agents. Since potassium iodide was administered to thousands of people in the Soviet Union as a result of the Chernobyl accident, we hope that useful data regarding its health effects will now become available.

Other emergency response items highlighted by the accident include the importance of effective procedures for relocating large population groups, protecting ground and other drinking water supplies, decontaminating land and facilities, and protective measures for minimizing radionuclide intake through food and other pathways.

The accident at Chernobyl reinforces a previous ACRS concern that the effects of an accident involving a large release of radioactive materials outside containment might negate safe habitation of the control room and other necessary facilities of the affected plant, or other units at a multiple-unit site.

Sincerely,

WKerr

William Kerr
Chairman

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

May 13, 1987

The Honorable Lando W. Zech, Jr.
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Zech:

SUBJECT: ACRS REPORT ON PROPOSED RESEARCH TO REDUCE SOURCE TERM UNCERTAINTY

During the 325th meeting of the Advisory Committee on Reactor Safeguards, May 7-9, 1987, we discussed a proposed research program for resolution of source term uncertainty areas as described in SECY 86-369, "Plan To Address Source Term Technical Uncertainty Areas." We also considered BNL report NUREG/CR-4883, an evaluation of this program by panels of experts sponsored by NRC. The ACRS Subcommittee on Severe Accidents considered this matter during a meeting on April 22, 1987. In our review, we had the benefit of discussions with the NRC Staff and the documents referenced.

We commend the expert panels for their expedited review and for their comments concerning some very complex phenomena. We agree generally with their findings and recommend that the Staff give careful consideration to their suggestions in planning the proposed research program.

We make the following additional observations:

(1) In our report dated June 10, 1986 in which we commented on NUREG-0956, "Reassessment of the Technical Bases for Estimating Source Terms," we recommended that the Staff attempt to quantify the uncertainties that were identified. The expert panels also noted that there are no quantitative estimates of the magnitude of the identified uncertainties. We agree with the panels that those planning the research programs need guidance as to which contributors to uncertainty are most important. To provide this guidance, the Staff should attempt not only to specify uncertainties in the descriptions of particular phenomena, but should also estimate their contribution to risk. There is also a need for an estimate of the level of uncertainty that is acceptable in making regulatory decisions. Although SECY 86-369 identifies areas of uncertainty, it does not indicate what level of uncertainty would be acceptable,

nor does it indicate how likely it is that the proposed research will reduce the uncertainty to an acceptable level.

(2) In the areas of steam explosions and hydrogen combustion, one of the panels recommended a reduction in research activities. For steam explosions within the vessel that lead to early containment failure, the consensus is that the conditional probability for such an event is very small (0.01), and thus need not be considered further. This panel further concluded that hydrogen combustion is reasonably well understood and that uncertainty in its understanding contributes relatively little uncertainty to estimates of source terms and risk. However, significant uncertainties do remain in regard to the generation of hydrogen during an accident. With the evidence now available to us, we agree with the panel's recommendation.

(3) A panel concluded that information needed to reduce the uncertainty in risk estimates for direct containment heating (DCH) will not be available within the next four or five years, even if a crash program is implemented. In light of this estimate, the panel recommended the exploration of plant changes (hardware or procedures) which would eliminate the sequence. The panel also recommended that the DCH experimental program be reorganized to show the effects of water and structural failure on DCH. We concur in both recommendations. In general, we conclude that the existing program is too narrowly focused. The program should be redirected to encompass a broader range of possible scenarios, including estimates of realistic mass flows from the vessel and possible vessel failure modes. The question of what is credible in the various situations must not be submerged in some large computer code, but should initially be sorted out by more straightforward and transparent physical arguments concerning the range of possibilities.

(4) There has been considerable discussion of the uncertainty associated with the chemical form of iodine, either volatile (elemental) or non-volatile (chemically bound as in CsI). After the TMI-2 accident, the absence of elemental iodine led some to conclude that the estimated risk should be reduced by a factor of as much as 100 from risks reported in WASH-1400, where it was assumed that all of the iodine was in elemental form. It is now reported that in studies conducted in the preparation of NUREG-0956, the difference in risk for volatile vs. non-volatile iodine is only about a factor of 3. A lesser priority should be assigned to research in this area.

(5) We observe that estimates of accident progression at key points in the core melt sequence depend on the prediction, using inadequately based computer codes, of such parameters as melt temperature and

time required for vessel melt-through. There appear to be significant uncertainties in the predictions of a number of these key parameters that tend to be masked by the codes. Since vessel penetration, core-concrete interactions, and the concurrent release of fission products, for example, are all very sensitive to melt temperature, we urge that efforts, including both experiments and independent calculations, be made to provide some independent and more transparent assessment of the behavior of key parameters. Comparison with another code embodying the same underlying assumptions is not sufficient.

(6) In light of the importance of containment behavior in determining the magnitude of the source term, we recommend that more attention be given to the identification and evaluation of other scenarios having the potential for leading to a large release of radioactive material.

Additional comments by ACRS Member Glenn A. Reed are presented below.

Sincerely,

WKerr

William Kerr
Chairman

<u>Additional Comments by ACRS Member Glenn A. Reed</u>

While I agree with the ACRS letter to reduce the research in described areas, I wish to focus on the panels' observation made as a "first suggestion" in the general conclusions that a prevention technique of "depressurization" (procedures and design) was important "to make the problem go away."

I recommend that research be increased and accelerated on the depressurization idea and that the research include application of depressurization as an alternative technique for core decay heat removal.

References:

1. U.S. Nuclear Regulatory Commission Staff Document, "Plan to Address Source Term Technical Uncertainty Areas," SECY-86-369, dated December 11, 1986
2. Brookhaven National Laboratory Report, "Review of Research on Uncertainties in Estimates of Source Terms From Severe Accidents in Nuclear Power Plants," NUREG/CR-4883, dated April 1987
3. U.S. Nuclear Regulatory Commission Report, "Reassessment of the Technical Bases for Estimating Source Terms," NUREG-0956, Draft Report for Comment, dated July 1985
4. U.S. Nuclear Regulatory Commission Report, "Reactor Safety Study -- An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," WASH-1400 (NUREG-75/014), dated October 1975
5. National Research Council Report, "Technical Aspects of Hydrogen Control and Combustion in Severe Light-Water Reactor Accidents," dated 1987
6. U.S. Nuclear Regulatory Commission Report, "A Review of the Current Understanding of the Potential for Containment Failure from In-Vessel Steam Explosions," NUREG-1116, dated June 1985.

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

June 9, 1987

The Honorable Lando W. Zech, Jr.
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Zech:

SUBJECT: ACRS REPORT ON PROPOSED GENERIC LETTER ON INDIVIDUAL PLANT EXAMINATIONS FOR SEVERE ACCIDENT VULNERABILITIES

During the 326th meeting of the Advisory Committee on Reactor Safeguards, June 4-6, 1987, we discussed a draft Generic Letter prepared by the NRC Staff as guidance for individual plant examinations (IPEs) for severe accident vulnerabilities. The IPEs are a part of an implementation plan for the Severe Accident Policy Statement. The ACRS Subcommittee on Severe Accidents considered this matter during meetings on December 19, 1986 and on May 28, 1987. In our review, we had the benefit of discussions with the NRC Staff and with representatives of the Industry Degraded Core Rulemaking (IDCOR) Program. We also had the benefit of the documents referenced.

The letter in its final form, accompanied by a panoply of supporting documents, is intended to provide guidance to nuclear power plant licensees in their performance of the individual plant examinations referred to in the Commission's Severe Accident Policy Statement (50 FR 32138, August 8, 1985). Specifically, the Policy Statement states:

> Accordingly, when NRC and Industry interactions on severe accident issues have progressed sufficiently to define the methods of analysis, the Commission plans to formulate an integrated systematic approach to an examination of each nuclear power plant now operating or under construction for possible significant risk contributors (sometimes called "outliers") that might be plant specific and might be missed absent a systematic search.

The NRC Staff finds that the following five options could satisfy the examination requirements, if appropriately supplemented:

(1) A PRA may be utilized, provided it is at least at Level II or Level III and it uses current methods and data.

(2) The IDCOR Individual Plant Evaluation Methodologies (IPEMs) may be used, provided the enhancements in the NRC Staff evaluation are applied. (The NRC Staff evaluation of the applicable IDCOR IPEM is included as an attachment to the generic letter.)

(3) A Level I PRA supplemented by an appropriately evaluated source term method may be applied.

(4) A simplified PRA which uses reduced systems models for the core damage analysis and sequence grouping for the containment performance analysis may be applied with an appropriate NRC approval.

(5) Another systematic examination method may be applied with prior NRC approval.

The NRC Staff requests documentation of the examination results, as follows:

(1) Certification that an IPE has been completed and documented as requested by the provisions contained in the generic letter.

(2) A listing of the dominant sequences leading either to core damage or to significant releases from containment and their frequencies for the plant, together with the screening criteria used to identify the sequences.

(3) Identification and listing of the main drivers, or leading contributors, to the predicted core damage frequency.

(4) Identification and listing of the main contributors to any predicted containment failure.

(5) A discussion of the potential areas of improvement identified in the plant examination which could reduce either the probability of severe accidents or the probability of large releases from severe accidents.

(6) A list of the most cost-effective potential improvements, including hardware changes as well as changes in procedures and training programs.

(7) An evaluation of the most promising improvements, disposition of those improvements, and an implementation schedule.

(8) Consistent with the assumptions made in the IPE, a description of organizational responsibilities related to severe accidents together with the steps taken to assure that personnel are

properly trained, appropriate procedures are in place, and diagnostic instruments and essential equipment will be available and will function where needed.

The PRA methods are relatively well specified from recent experience, at least for internal events up to the core damage stage. With regard to the IDCOR IPEMs, the NRC Staff has provided evaluations which lead to a large number of recommended modifications and additions that are needed to make the IDCOR IPEM option acceptable. The ACRS generally supports the NRC Staff's evaluations of the IDCOR IPEMs.

We recognize that formulating guidance for an individual plant examination is a formidable task. We commend the NRC Staff for the progress that has been made, and for the cooperation with industry, through the IDCOR program, that has produced a significant contribution to the effort. However, we believe that the proposed guidance that has been prepared is deficient in a number of areas, and that unless it is improved before licensees are required to design a program and perform an examination, a number of important objectives of the program are unlikely to be achieved.

The suggested approach to plant analysis is divided into two segments called "front end" (i.e., the description of an hypothesized sequence from initiation to the beginning of severe core damage) and "back end" (i.e., from the onset of severe core damage to release of radioactive material from containment). The guidance emphasizes that the two segments are not altogether independent. However, because the onset of severe core damage or core melt has become something of a milestone in many PRAs, this is probably a reasonable division. The guidance given for the front end analysis in the current draft is much more detailed, and would be much easier for an inexperienced group to follow, than is the guidance for the back end which deals primarily with post-core-melt severe accident progression and containment performance. We believe that the guidance given, and the methods suggested, can provide a reasonable basis for a search for vulnerabilities in the pre-core-melt or preventive part of postulated sequences. However, the so-called guidance on containment system performance analysis, especially that part that deals with PWRs, appears to be a rather hurriedly assembled discussion of some of the problems and uncertainties likely to be encountered in the analysis of containment performance, with very little guidance on how to perform a search for vulnerabilities.

We recognize and support the NRC Staff's effort not to be overly prescriptive. Furthermore, the contrast between the guidance given for the front end and the back end analyses reflects, to some extent, the relative state of development of information needed to perform an analysis of reactor system performance, compared to that needed to describe containment system performance. Nevertheless, it is our judgment that if licensees, especially those with limited PRA experience, are faced with guidance on containment performance analysis as ambiguous as that in the current draft, they will be so mystified that they will have no recourse but to retain an outside group to carry out the analysis. They will thereby miss one of the more important benefits of the IPE, that of becoming familiar enough with system performance to be able to recognize vulnerabilities in their plants, and of becoming aware of expected system performance in a severe accident. The guidance on containment analysis should be improved before the letter is released.

We also believe that not enough guidance is given as to goals and objectives of the examination. The draft letter, in describing the Commission's Policy Statement, identifies the "overall goals of the policy" as "(1) to reduce the probability of a severe accident, and (2) should a severe accident occur, to mitigate, to the extent possible, its consequences to the public." It identifies the purpose of the examination as providing "the basis for a utility's appreciation of severe accident behavior, recognition of the role of prevention and mitigation systems and procedures, and the development of an accident management scheme." A licensee must also, having discovered possible vulnerabilities, identify potential areas for improvement, suggest corrective actions to achieve improvement, decide which improvements he thinks should be implemented (if any), discuss the decision not to make those judged inappropriate, and give a schedule for effecting those changes that are planned: all of this before the examination has been reviewed by the NRC Staff. The licensee is also asked to develop an organized approach, including training to deal with many severe core damage accidents.

Vulnerabilities are not defined, either qualitatively or quantitatively (except perhaps by inference from some of the material referenced in the letter), nor is there guidance as to the amount and kind of improvement that the NRC Staff will find acceptable. The reason given for not providing further guidance is that there are no objective standards, that each licensee must make a decision for himself as to the changes that are appropriate. However, the reviewing NRC Staff will need to have some criteria to provide a basis for review. It would save everyone a considerable amount of thrashing about if more guidance could be given as to criteria to be used in determining the adequacy of the IPEs.

From our discussions with the NRC Staff, we have concluded that the projected scope of the review described by the draft letter may be too ambitious. Based on our earlier discussions with the NRC Staff, we had concluded that the IPEs were to be performed to look for "outlier" plants, i.e. plants with features, procedures or other operating characteristics which produced risks unexpectedly high compared with those of the general population. It appears, however, that the program currently envisioned is one which attempts to establish a profile of core melt frequency and containment performance (described at least semi-quantitatively, if not quantitatively) for each operating plant, and then (possibly) attempts to reduce plant risk to some unspecified level, not necessarily the same for each plant, by requiring plant or other modifications which reduce the contribution from some selected population of risk contributors. It would also lead to the beginning of a risk management program at each plant. Although there may be merit in this approach, we question whether many of these tasks are suitable for individual initiatives; rather they would need the efforts of appropriate new owners' groups, and NRC Staff guidance would have to be improved.

The guidance provided makes it clear that analyses of severe accident sequences initiated by external events and by sabotage are not requested at this time. Analyses for external initiators will be required later. Nevertheless, it would be helpful to give at least some guidance at this time as to what is likely to be asked for later on, especially since one option given a licensee is to perform a PRA which considers external events.

In light of both the difficulty and the importance of the IPEs, we recommend that instead of the approach proposed in the draft letter, which has all operating plants begin the review immediately, the NRC Staff arrange trial reviews of several plants to be carried out cooperatively with licensees in somewhat the same way that the Systematic Evaluation Program (SEP) reviews were performed. Although part of the the review process developed by IDCOR has been exercised by them on several plants, the NRC Staff's view is that IDCOR's treatment of containment performance does not consider several important safety-related questions. Furthermore, for most of the plants reviewed by IDCOR, a more extensive PRA existed. Such reviews provide a useful reference. However, it would be valuable to perform reviews for a few plants that do not have PRAs. If this were done cooperatively by the NRC Staff and the licensees, it could provide additional information on the application of non-PRA approaches, and could also serve as a tool for development of more sharply focused guidance for later IPEs.

Sincerely,

WKerr

William Kerr
Chairman

References:
1. Letter from T. P. Speis, NRC, to W. Kerr, ACRS, Subject: Documentation Necessary for the Initiation of the Severe Accident Policy Implementation, dated May 15, 1987.
2. Letter from T. P. Speis, NRC, to W. Kerr, ACRS, Subject: Documentation Necessary for the Initiation of the Severe Accident Policy Implementation, dated May 22, 1987.
3. U.S. Nuclear Regulatory Commission, "NRC Policy on Future Reactor Designs - Decisions on Severe Accident Issues in Nuclear Power Plant Regulation," USNRC Report, NUREG-1070, dated July 1985.
4. IDCOR Program Reports (IDCOR-IPEMs):
   (a) Technical Report 86.3A1, "Individual Plant Evaluation Methodology for Pressurized Water Reactors," April 1987,
   (b) Technical Report 86.3A2, "IPE Source Term Methodology for PWRs," March 1987,
   (c) Technical Report 86.3B1, "Individual Plant Evaluation Methodology for Boiling Water Reactors," Volumes I and II, April 1987,
   (d) Technical Report 86.3B2, "IPE Source Term Methodology for BWRs," March 1987.
5. Brookhaven National Laboratory Reports (Draft), "Prevention and Mitigation of Severe Accidents," NUREG/CR-4920, dated March 1987
   (a) Volume 1, "BWR, Mark I Design,"
   (b) Volume 2, "BWR, Mark II Design,"
   (c) Volume 3, "BWR, Mark III Design,"
   (d) Volume 4, "PWR, Large Volume Containment Design," and
   (e) Volume 5, "PWR, Ice Condenser Containment Design."

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

September 16, 1987

The Honorable Lando W. Zech, Jr.
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Zech:

SUBJECT: ACRS COMMENTS ON CODE SCALING, APPLICABILITY AND UNCERTAINTY METHODOLOGY FOR DETERMINATION OF UNCERTAINTY ASSOCIATED WITH THE USE OF REALISTIC ECCS EVALUATION MODELS

During the 329th meeting of the Advisory Committee on Reactor Safeguards, September 10-12, 1987, we reviewed the methodology developed by the NRC Office of Nuclear Regulatory Research for determination of the overall uncertainty associated with the use of realistic models, including related computer codes, for the calculation of thermal-hydraulic phenomena associated with loss of coolant accidents (LOCAs). In our review, we had the benefit of discussions with representatives of the Office of Nuclear Regulatory Research (RES) and the Office of Nuclear Reactor Regulation (NRR). Subcommittee meetings during which this topic was discussed were held on April 29-30, 1986, August 28, 1986, April 29-30, 1987, and August 4, 1987. We also had the benefit of the documents referenced.

A recently proposed revision to the ECCS Rule (10 CFR 50.46 and Appendix K) will permit use of realistic or "best estimate" methods in demonstrating that a peak cladding temperature (PCT) of 2200°F will not be exceeded during a LOCA. This is in contrast to the original version of the rule which insisted on the use of a number of conservative assumptions which were believed to provide an overestimate of PCT large enough to account for uncertainties. With the new rule change, a licensee may demonstrate that the calculated PCT, when adjusted with an appropriate allowance for overall uncertainty, has an estimated 95% probability of not exceeding 2200°F. In our September 16, 1986 letter to you commenting on the proposed ECCS Rule, we noted the following:

> "The acceptability of realistic evaluation models rests on the development of satisfactory methodology for determination of the overall uncertainty. Most of the development work needed here is either ongoing or planned by the Office of Nuclear Regulatory Research. We recommend that the methodology used to evaluate uncertainty be subjected to peer review. We also wish to review this work."

RES has developed a method for quantifying uncertainty in PCT which it refers to as the Code Scaling, Applicability and Uncertainty (CSAU) evaluation methodology. CSAU is designed to address uncertainties in the capability of a code to extrapolate small-scale test data to full scale, to correctly assess a particular sequence of events, and to account for variability in important parameters. The focus of CSAU is on the important thermal-hydraulic processes with detailed attention given only to those processes which contribute importantly to overall uncertainty. The end product of the CSAU method is an estimate of the total uncertainty associated with the calculation of a key parameter (e.g., PCT) by a given realistic thermal-hydraulic code for a particular plant and a particular accident transient.

It must be recognized that absent an abundance of full-scale LWR plant transient data, it is necessary to rely substantially on engineering judgement in lieu of a rigorous statistical analysis. The CSAU methodology systematizes the application of this judgment for the derivation of a quantitative allowance for uncertainty.

We believe that the CSAU method proposed by RES offers an acceptable means to estimate uncertainty associated with the use of realistic codes. However, we wish to note the following:

- The CSAU methodology has not yet been tested over a wide range of applications. Currently, RES is in the process of demonstrating the applicability of the method by using it to determine the uncertainties resulting from a large break LOCA calculation using the TRAC PF1/MOD-1 code. While it appears that CSAU will be successfully applied to TRAC, we recommend that RES complete an adequate evaluation before the methodology is judged acceptable for use in regulatory actions.

- Before CSAU can be applied to a given code, complete documentation (e.g., code manual, model and correlation quality assurance document, and assessment reports) is necessary. In the past, such thorough documentation has not always been available for licensing codes. We recommend that steps be taken to ensure that future development of codes for licensing activities be performed in a manner that ensures completion and availability of needed documentation before the code is released.

- The codes used to analyze thermal-hydraulic behavior are very large and complex. Validity of calculated results is dependent on the competence of the code user and the way in which the code is used. For CSAU to be effective, the code developers, assessors and users must use the code consistently. We recommend the NRC Staff take the necessary steps to ensure that proper controls are established.

- In order to ensure the ultimate success of the method, we believe it is necessary for RES to direct its experimental thermal-hydraulic programs appropriately to the needs of CSAU. These experimental programs include the MIST, 2D/3D, and ROSA-IV cooperative efforts.

- We wish to caution that use of the CSAU method for regulatory applications will require the maintenance of an ongoing high level of competence and experience on the part of the NRC Staff members. We suggest that the NRR call upon RES for such support as necessary.

We are encouraged by the move toward the use of realistic calculations for ECCS/LOCA phenomena. We intend to follow the progress of this effort closely, and we wish to be kept informed.

Additional comments by ACRS Member Harold W. Lewis are presented below.

Sincerely,

WKerr

William Kerr
Chairman

Additional Comments by ACRS Member Harold W. Lewis

I support the Committee's letter, but do wish to add some cautionary notes about the misuse of some familiar words, which can lead to potential misuse of the CSAU (so-called) methodology.

To begin with, I support the move to "realistic" evaluations, since I believe that all evaluations should be made as honestly and realistically as possible, after which regulatory conservatism can be applied cleanly and openly. That is the thrust of this effort, and is fine. Unfortunately, however, the words "best estimate" are often used interchangeably with "realistic" to describe calculational techniques, and that is an error. To a statistician, a best estimate is an estimate taken from the top of a probability distribution, and that is simply a different idea. This is not sophistry, since the misunderstanding of words that have established technical meanings can lead to incorrect calculations. To call an apple an orange does not make it one.

We were also briefed about a set of calculations in which parameters and assumptions were varied to provide a feel for the sensitivity of the results to the specific assumptions made. That is a reasonable way to learn about the sensitivity, but is not a way to learn about the "uncertainty" in the result, as any statistician would understand the word uncertainty. Statistical uncertainty in its simplest form is based on the concept of random sampling from a population of known characteristics but unknown parameters. In that case, one can learn the uncertainty in an estimate of a parameter by studying the variance in a set of measurements, but that is not the situation here, where the variance in the results bears no relation whatever to any uncertainty, in any credible statistical sense. The only reason for saying this is that in the familiar case of a normal distribution of sample measurements, one can estimate the uncertainty from the variance, and thereby estimate the probability that the mean of the measurements differs from the true value by any ratio. One can also estimate "confidence levels," but that is another saga.

None of that is true here, and this is again not sophistry. In particular, the draft Regulatory Guide supporting the proposed rule has statements about the "95% probability limit," "confidence level," and such things, and even states that the "use of two standard deviations for evaluating the 95% probability level is acceptable." None of this is possible within the framework described, and simply reflects confusion on the part of the Staff about fundamental statistical concepts.

I still support the letter and the program, since it is a major step forward, but repeat a recommendation I have made many times: the NRC would benefit greatly by hiring a few good statisticians. One cannot do competent safety analysis in the presence of uncertainty (popular use of the word) without doing the statistics carefully.

References:

1. U.S. Nuclear Regulatory Commission, Proposed Rule, "Emergency Core Cooling Systems, Revisions to Acceptance Criteria," February 26, 1987.
2. U.S. Nuclear Regulatory Commission, "Request for Comments on Draft Regulatory Guide, 'Best Estimate Calculations of Emergency Core Cooling System Performance,'" March 1987.
3. U.S. Nuclear Regulatory Commission, NUREG-1230, "Compendium of ECCS Research for Realistic LOCA Analysis," April 1987.

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

September 16, 1987

The Honorable Lando W. Zech, Jr.
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Zech:

SUBJECT: ACRS COMMENTS ON DEVELOPMENTS IN EMERGENCY PLANNING

During the 329th meeting of the ACRS, September 10-12, 1987, we met with representatives of the New York Power Authority, the Oak Ridge Associated Universities, and the Sandia National Laboratories to discuss preliminary analyses of the benefits of various measures taken to protect the population in case of a major accident at a nuclear power plant. Representatives of the NRC Staff took part in the discussion. This matter was also the subject of a meeting of our Subcommittee on Occupational and Environmental Radiation Protection Systems held on June 22-23, 1987.

Studies reported by these groups indicate that sheltering, followed by monitoring of radiation exposure rates and relocation of populations from affected high radiation areas, within 4 to 8 hours after an accident, yields predictions for the number of prompt fatalities lower than those estimated to be provided by the evacuation expected under current decision-making practices. This was the statistical result of a wide range of accident scenarios. Since the number of people, the distance they would need to be moved, and the disruptive impact of the sheltering-relocation approach would normally be less than those for the immediate evacuation approach, we believe that the NRC Staff should be asked to conduct an independent and prompt assessment of these findings. Should this assessment confirm the reported observations, there appears to be reason for emphasizing sheltering, where appropriate, in nuclear emergency response.

Sincerely,

WKerr

William Kerr
Chairman

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

February 10, 1987

Mr. Victor Stello, Jr.
Executive Director for Operations
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Stello:

SUBJECT: ACRS COMMENTS ON TESTING OF CHARCOAL ADSORPTION CAPACITY

During the 322nd meeting of the Advisory Committee on Reactor Safeguards, February 5-7, 1987, we discussed the capability for testing charcoal adsorption capacity in filters used at nuclear power plants.

Current Technical Specifications require periodic testing of the charcoal in adsorption units designed to control releases of airborne radioiodine from nuclear power plants. However, "round robin" tests have shown that most commercial laboratories, both in the U. S. and abroad, lack the capability to determine the adsorption capacities of filter charcoals on an accurate and reliable basis. Although the NRC has supported research on this problem, current NRC plans are to terminate this support, based on the expectation that industry will assume responsibility for continuing this research.

In connection with NRC termination of this work, we believe that the industry group that is to assume responsibility to continue this work should be identified and assurances made that the program will be pursued to a successful completion.

Sincerely,

WKerr

William Kerr
Chairman

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

July 16, 1987

The Honorable Lando W. Zech, Jr.
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Zech:

SUBJECT: ACRS COMMENTS ON RESEARCH INTO CONTINUOUS CONTAINMENT LEAKAGE MONITORING

In some of our recent discussions, the concept of continuous containment leakage monitoring has resurfaced. We believe there may be merit in this concept for reducing the risk of exposure to the public and plant operators in severe accident situations.

We recommend an investigation of continuous containment leakage monitoring to see if it can be helpful in risk reduction and if it is cost beneficial.

Sincerely,

WKerr

William Kerr
Chairman

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

August 10, 1987

The Honorable Lando W. Zech, Jr.
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Zech:

SUBJECT: PRELIMINARY ACRS VIEWS ON FIRE RISK RESEARCH SCOPING STUDY

During the 328th meeting of the ACRS, August 6-8, 1987, we discussed the scope, direction, and current status of the Fire Risk Scoping Study being performed by Sandia National Laboratories (SNL) for the NRC. The ACRS Subcommittee on Auxiliary Systems also discussed this matter at a meeting on July 23, 1987. In our review, we had the benefit of discussions with representatives of the Office of Nuclear Regulatory Research (RES) and SNL.

In the ACRS report of February 19, 1986 to the Congress on the FY 1987 NRC safety research program, and also in its June 11, 1986 report to the Commission on the FY 1988 research program and budget, the Committee expressed concern about terminating the fire protection research program at the end of FY 1986 and recommended that funding for research in this area be restored. The RES response to this concern was the initiation of a scoping study on the risk of fires to determine if further fire-related research is warranted. This study is to utilize results of completed research and the fire risk analysis which is now nearing completion for the LaSalle County Station nuclear plant.

In the ACRS letter of July 16, 1986, the Committee expressed concern about the loss of program information and momentum that would result from premature termination of ongoing fire-related research activities while awaiting the results of the scoping study. The Committee noted that termination of the needed research would be a serious loss, and would be costly to reconstitute later.

Although the Commission agreed with the ACRS on the importance of fire protection research, it did not restore the funding. However, it did direct the Staff to work closely with the Committee to assess further research needs and to consider the priority that should be assigned to fire protection research. A good relationship was established and efforts are proceeding on schedule to assign a priority to possible research needs.

Various tasks are now progressing and the work is scheduled for completion in December 1987. The study includes identification of various potential fire-related issues, including those cited by the ACRS, and an assessment of the risk significance of such issues. The risk considerations include an assessment of uncertainties in various previous probabilistic risk assessments (PRAs) and a requantification of PRA fire scenarios. The final task will deal with the completeness of 10 CFR Part 50, Appendix R and other fire-associated regulatory requirements as they may relate to potential fire issues. Although the scoping study is still under way, we believe that the Commission may wish to be informed of our preliminary views which follow.

The main objective of the Fire Risk Scoping Study is to assess the risk significance and dominant sources of uncertainty associated with fire risk issues, with a final goal of assigning an appropriate priority for possible fire-related research. We believe that the study is progressing satisfactorily toward this goal and is targeting the various concerns expressed by the Committee. The scope appears to be providing a needed and timely basis for determining priorities. We plan to review and issue comments on the final results of the scoping study. The recommended priority and the technical aspects of any proposed fire research program, including interim or long-range budgetary needs, will be discussed at that time.

Sincerely,

WKerr

William Kerr
Chairman

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

March 10, 1987

MEMORANDUM FOR: Mr. Victor Stello, Jr.
Executive Director for Operations

FROM: Mr. Raymond F. Fraley
Executive Director, ACRS

Raymond F. Fraley

SUBJECT: ACRS COMMENT ON PRIORITIZATION OF GENERIC ISSUE 61: "SRV DISCHARGE LINE BREAK INSIDE THE WETWELL AIR-SPACE OF BWR MARK I AND MARK II CONTAINMENTS"

During the Committees' review of the fourth group of generic issues at the 319th meeting of the ACRS (November 6-8, 1986), the members deferred comment on the subject issue pending additional review. The Committee has now completed review of this matter and its comments are attached.

As the Committee has "agreed with comment" on Generic Issue 61, the members have requested a written response from the NRC Staff to their comments.

Please note that the attached comments recommend the evaluation of a potential new generic issue.

Attachment: As Stated

Comment on Priority Ranking for
Generic Issue 61
"Agrees With Comments"

Generic Issue No:	61
Title:	SRV Discharge Line Break Inside the Wetwell Airspace of BWR Mark I and Mark II Containments
Priority Ranking Proposed by the NRC Staff:	DROP
ACRS Comments:	The ACRS agrees with the proposed priority ranking for this issue. However, during our consideration of this issue, a related concern arose: the issue of potential containment overpressurization given a steam and/or large coolant release in the drywell and bypass to the wetwell airspace through a stuck- open wetwell-to-drywell vacuum breaker (air-return valve). Our inquiries to NRR indicate that this particular accident scenario has never been directly addressed by the Staff. We recommend that NRR evaluate this item as a potential generic issue to ensure its appropriate resolution.

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

June 9, 1987

The Honorable Lando W. Zech, Jr.
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Zech:

SUBJECT: ACRS COMMENTS ON THE NRC STAFF PROPOSAL FOR THE RESOLUTION OF USI A-44, "STATION BLACKOUT"

During the 326th meeting of the ACRS, June 4-6, 1987, and in our 325th meeting on May 7-9, 1987, we discussed the resolution of USI A-44, "Station Blackout," that is being proposed by the NRC Staff. We also discussed the Nuclear Utility Management and Resources Committee (NUMARC) initiatives directed at reducing the risk from "Station Blackout." A Subcommittee meeting was also held to discuss this issue with the NRC Staff on May 6, 1987. During these meetings, we had the benefit of presentations by representatives of the NRC Staff and NUMARC. We also had the benefit of the documents referenced.

Since March 30, 1982, members of the ACRS have considered and discussed this issue at nine meetings, and offered comments to the Executive Director for Operations in letters dated July 13, 1983 and March 12, 1985. The ACRS has been generally receptive to and supportive of the Staff's efforts in seeking resolution of the issue.

We consider the proposed resolution of USI A-44, "Station Blackout," to be workable, and we commend the Staff for its efforts. However, we do not recommend issuance of the final rule at this time.

We believe that the NUMARC initiatives may be a viable alternative for dealing with this issue on an expeditious schedule and may require the least expenditure of resources on the part of the industry. We believe that the electric utility industry has a strong incentive to deal with "Station Blackout."

One shortcoming of the proposed NUMARC initiatives is the absence of a requirement for any assessment of a plant's ability to cope with station blackout for a specified length of time. A letter from NUMARC has advised us that they are developing a methodology to do this, but that industry-wide agreement will have to be obtained. They expect that the development of their initiatives will be substantially completed by September of this year.

We recommend that the Staff continue to work with NUMARC on the technical aspects of the NUMARC efforts. If by September of this year it is determined by the Staff that the NUMARC initiatives will not be effective or timely in reducing the risk from "Station Blackout" to acceptable levels, or that the NUMARC initiatives will be unduly difficult to evaluate on a plant-to-plant basis, we then recommend issuance of the final rule.

Additional remarks by ACRS Members Glenn A. Reed and Charles J. Wylie are presented below.

Sincerely,

WKerr

William Kerr
Chairman

Additional Remarks by ACRS Members Glenn A. Reed and Charles J. Wylie

We believe the NRC Staff has done a commendable job in bringing A-44 to resolution. However, we continue to support two previous ACRS letters (July 13, 1983 and March 12, 1985) recommending in part that A-44 implementation should be integrated with A-45, "Shutdown Decay Heat Removal Requirements." Unfortunately A-45 has not arrived at the same status, and the NRC Staff wishes to proceed now with a rule and guide on station blackout which deal with A-44 only. But, the root issue is not station blackout but rather decay heat removal to limit core melt risk to an appropriate level.

We do not consider it in the best interest of nuclear safety to proceed now with an NRC rule and guide on station blackout, which could compromise future desirable and more effective action for decay heat removal. Since it appears that NUMARC-Nuclear Utilities Group on Station Blackout (NUGSBO) has also been moving forward with an industry effort, and since the electric utilities should have premiere capabilities to upgrade vulnerabilities to station electrical blackout, we recommend NUMARC-NUGSBO carry the ball, with NRC Staff interfacing and monitoring -- but without an NRC rule. This arrangement would leave the NRC uncompromised to act appropriately on A-45 when its resolution is completed. In our opinion there may be some outlier units for which it is more preferable to focus and expend funds on the root issue of decay heat removal without diverting effort to station blackout; and such focusing may be more harmonious with the backfit rule.

References:

1. U.S. Nuclear Regulatory Commission, Federal Register Notice (51 FR 9829) for the proposed Station Blackout Rule (10 CFR 50.63), published on March 21, 1986.
2. U.S. Nuclear Regulatory Guide on "Station Blackout," dated March 30, 1987.
3. U.S. Nuclear Regulatory Commission, NUREG-1109, "Regulatory/Backfit Analysis for the Resolution of Unresolved Safety Issue A-44," submitted March 30, 1987.
4. U.S. Nuclear Regulatory Commission, NUREG-1032, "Evaluation of Station Blackout Accidents at Nuclear Power Plants," draft, submitted April 16, 1987.
5. U.S. Nuclear Regulatory Commission, NUREG/CR-3226, "Station Blackout Accident Analyses," dated May 1983.
6. U.S. Nuclear Regulatory Commission, NUREG/CR-2989, "Reliability of Emergency AC Power Systems at Nuclear Power Plants, dated July 1983.

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

October 15, 1987

The Honorable Lando W. Zech, Jr.
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Chairman Zech:

SUBJECT: ACRS COMMENTS ON THE PROPOSED RESOLUTION FOR GENERIC ISSUE 124, "AUXILIARY FEEDWATER SYSTEM RELIABILITY"

During the 330th meeting of the Advisory Committee on Reactor Safeguards, October 8-10, 1987, we completed discussion of the status of a resolution for Generic Issue 124 (GI-124) concerning the reliability of auxiliary feedwater (AFW) systems in seven particular plants. The Committee previously met with representatives of the NRC Office of Nuclear Reactor Regulation (NRR) and the Office of Nuclear Regulatory Research during our 329th meeting, September 10-12, 1987. This matter was also discussed during a meeting of the Decay Heat Removal Systems Subcommittee on August 5, 1987. We reviewed the beginning of this work about a year ago and commented in a letter dated September 17, 1986 to the Executive Director for Operations. We also had the benefit of the documents referenced.

GI-124 addresses concerns about the adequacy of AFW systems in a particular set of seven older PWR plants. These plants had been singled out for generic attention in a screening study of AFW system reliability several years ago. It was believed that this group of plants deserved special attention in advance of the more general review of the reliability of decay heat removal (which includes the issue of AFW reliability) in all plants being evaluated in the Unresolved Safety Issue A-45 (USI A-45) program.

Each of the seven plants has a two-train AFW system estimated, at the time of the screening, to have an unreliability greater than 10^{-4} per demand. Other "two-train" plants, which had estimated unreliabilities less than 10^{-4} per demand were not included in the group of seven plants.

Our 1986 letter was critical of the proposed program plan because it failed to identify objective criteria by which reliability or effectiveness of AFW systems were to be judged. The NRC Staff responded by asking that we wait until the initial plant reviews were available and then reconsider whether we agreed with their approach to resolution as put into practice.

We have now reviewed the initial plant evaluations and our objection to the process remains. As we understand the resolution process, it is to consist of seven plant-specific evaluations and negotiated settlement packages, rather than a general solution. Each evaluation starts with an inspection and review of the design and operation of a plant's AFW system by an NRR team. The inspection and review identifies "negative features" in design, operation, or maintenance and calls these to the attention of the licensee. It is then, apparently, the intent to correct or otherwise resolve these negative features to the mutual satisfaction of the licensee and NRR.

Our objection to this approach has two main points:

(1) The quantitative criterion (unreliability greater than 10^{-4} per demand) by which the seven plants were originally singled out as requiring special attention has been rejected by the NRC Staff as too "crude" to be used in measuring the adequacy of proposed AFW improvements. This calls into question the original selection process. It becomes unclear whether there really is a generic issue regarding AFW reliability in a certain subset of plants and, if there is, why these particular seven plants are in the subset of concern.

(2) The NRC Staff has not specified an objective standard by which it intends to judge whether possible improvements to the AFW systems in these plants are adequate. Instead, NRC Staff teams will review each AFW system in detail, react to what they find, and negotiate improvements with the licensees. We believe this approach represents a serious misallocation of responsibility and resources between regulators and the regulated industry. It is a mistake that should be corrected in this instance and in other regulatory activities as necessary.

We will expand on each of these two points below.

If the screening analysis used to identify this subset of seven plants as having a unique problem is now considered to be seriously flawed, then we believe the whole basis for GI-124 is invalid. It may be most appropriate to drop this issue and to concentrate Staff resources on the resolution of USI A-45.

If GI-124 is to be continued, the conditions important to AFW reliability should be considered more explicitly in the resolution. From a risk perspective, the minimum acceptable AFW reliability is related to the expected challenge or demand frequency on the system. For example, if the main feedwater (MFW) system in a plant is capable of maintaining stable flow to the steam generators for an extended period following a reactor and turbine trip, then the reliability requirement on AFW might be lower than otherwise deemed acceptable for a plant without this capability. Of course, if trips of the MFW system itself are a main cause of demand for AFW, this advantage might be unimportant. As another condition, if there is a strong capability for primary bleed and feed heat removal in a plant, again the reliability requirement on AFW might be lower than otherwise considered acceptable.

It appears to us that the plant reviews conducted so far have been done competently by experienced and capable review teams. Negative features identified have been real and practical issues, but often of rather minor individual significance. Some more significant design or operational problems have also been identified. If all or most of the individual issues are corrected or improved, there is little doubt that AFW reliability will be somewhat improved at each of the plants. This is a subjective judgment on our part because NRR has furnished no quantitative estimates of the incremental risk associated with each negative observation -- nor with their sum.

Our objection to this approach for resolution of GI-124 is not that the process itself entirely lacks merit, but that it is inappropriate for NRC to resolve a generic safety issue in this manner. Inspection and review of the sort described to us should be carried out in-house by the utility-licensee or by an industry organization. The NRC should better use its own resources by providing the licensees with some objective definition of the AFW reliability it believes is necessary.

For example, if an unreliability for AFW greater than 10^{-4} per demand is judged by the NRC to be inconsistent with its overall intent in regulating nuclear power, then the resolution of GI-124 could require a good faith effort on the part of licensees to estimate the unreliability of the system in each plant. This would be followed by licensee-initiated improvement of the AFW system sufficient to meet that requirement. If the NRC believes analytical methods are not well enough developed to specify this sort of quantitative limit on unreliability, then it might instead want to specify a deterministic requirement, e.g., that two-train AFW systems are acceptable only if they incorporate certain favorable attributes or a diverse system for decay heat removal. But, the NRC must then have the resolve to define these necessary attributes in an understandable way and not resort to a reactive ("bring me a rock") style of regulation.

We recognize that the development of an appropriate objective criterion for AFW reliability is, or may be, a difficult task. However, diversion of the engineering resources of NRR to work that is more properly carried out by industry, such as the aforementioned inspection and review teams, only delays addressing the difficulty and may preclude development of a truly generic resolution that is both sound and has long-term utility.

Sincerely,

W Kerr

William Kerr
Chairman

References:
1. U.S. Nuclear Regulatory Commission, "Safety Evaluation by the Office of Nuclear Reactor Regulation, Auxiliary Feedwater System Reliability (Generic Issue 124) With Respect to Prairie Island Nuclear Generating Plant Unit Nos. 1 and 2," transmitted by letter from George Lear, Division of PWR Licensing-A, Office of Nuclear Reactor Regulation, to Dave M. Musolf, Northern States Power, dated November 26, 1986.
2. U.S. Nuclear Regulatory Commission, "Safety Evaluation of the Auxiliary Feedwater System (Generic Issue 124) With Respect to Arkansas Nuclear One Generating Plant Unit 2," transmitted by memorandum from Eric S. Beckjord, Office of Nuclear Regulatory Research, to Thomas E. Murley, Office of Nuclear Reactor Regulation, dated July 13, 1987.

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

July 15, 1987

Mr. Victor Stello, Jr.
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Stello:

SUBJECT: ACRS COMMENTS ON LICENSEE EVENT REPORTS PERTAINING TO CONTROL ROOM HABITABILITY

For a number of years, members of the Advisory Committee on Reactor Safeguards, supported by ACRS Fellows, have examined Licensee Event Reports (LERs) pertaining to air cleaning, ventilating, and monitoring systems at commercial nuclear power plants. Several of the more recent of these studies have concentrated on LERs specifically pertaining to control room habitability.

The latest of these studies, which covered the three-year period from 1984 through 1986, has revealed the following information:

1. On an annual basis, from 3% to 8% of all LERs pertained to systems related to control room habitability. For the three-year period, 7% of all LERs were in this category. This represented a total of over 500 LERs.

2. Of the LERs in this category, 61% were due to problems involving air monitors. Of these, 55% were due to problems with radiation monitors and 29% were due to problems with chlorine monitors.

Most of these events were reported as LERs because malfunctions of the monitoring equipment led to actuations of the control room emergency ventilation system. The large number of LERs in this category indicates a need to address attention to their origin and the need for corrective action. Such events almost certainly reflect a lack of reliability on the part of certain types of air monitoring equipment.

Several approaches may be useful in planning corrective action. Although malfunctions of chlorine monitors account for a significant percentage of the cited LERs, data for the past several years indicate essentially no problems with these types of monitors at certain nuclear power plants. It might be beneficial to determine whether such monitors are in use in these plants and, if so, what type they are and how they are maintained and operated. Such information could be useful in resolving some of the problems observed at other plants.

A second approach might be for the NRC Staff to consider encouraging all nuclear power plant licensees to adopt the provisions of the current Standard Technical Specifications which specify a time limit within which a defective air monitor would have to be repaired and placed back into service. Such a requirement would help make the management at all plants aware of the need to purchase and install reliable air monitoring equipment and to maintain it in proper working order.

Sincerely,

W Kerr

William Kerr
Chairman

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

July 15, 1987

Mr. Victor Stello, Jr.
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Stello:

SUBJECT: ACRS COMMENTS ON THE EMBRITTLEMENT OF STRUCTURAL STEEL

Surveillance samples of steel used in the pressure vessel of the High Flux Isotope Reactor (HFIR) at the Oak Ridge National Laboratory recently have shown that the nil-ductility transition (NDT) temperature of steel irradiated slowly at $120^{\circ}F$ can rise much more rapidly with exposure to fast neutrons than would be expected from the available experimental work obtained in test reactors. This appears to be due to two causes:

- a flux rate effect (A lower fast neutron flux embrittles more than the same fluence accumulated at a much higher flux in test reactors.)

- the difference in temperature ($550^{\circ}F$ for commercial reactor pressure vessels vs. $120^{\circ}F$ for the HFIR)

This has led to a significant shift in the NDT of the steel at a fast neutron fluence lower by roughly a factor of 20 than that predicted by the correlations used in the past for low temperature irradiations. This acceleration is independent of the copper content of the material. This suggests that steel structures outside the pressure vessel in commercial nuclear power plants may have embrittled where such behavior was not expected. We believe it would be prudent for the NRC to do the following:

1. Determine if the brittle failure of any structural steel component near the outside of the primary pressure boundary would have safety significance.

2. Determine, using the low temperature irradiation data now available from test reactors, whether an increase in the fast neutron fluence by a factor of 10-100 would be predicted to give brittle behavior in these components.

3. Implement a research program which would assemble better information on the rate of shift of the NDT of structural steels in

commercial nuclear power plants at these lower rates and temperatures.

4. Include consideration of the accelerated shift in NDT as part of the evaluation of structures in the program on plant aging.

Sincerely,

W Kerr

William Kerr
Chairman

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

December 8, 1987

Mr. Victor Stello, Jr.
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Stello:

SUBJECT: ACRS COMMENTS ON MEMORANDUM FROM VICTOR STELLO, JR., EDO, DATED OCTOBER 7, 1987 REGARDING THE EMBRITTLEMENT OF STRUCTURAL STEEL

We are concerned and perplexed by your memorandum of October 7 (referenced). There you conclude that, "the neutron shield tanks and support structures do not appear to pose any safety problems. The embrittlement can be conservatively predicted as an increase in transition temperature of the steel of as much as 400°F." You support your conclusion with the statement, "These structures are in compression, so even with a 0.2 g earthquake, the tensile stresses generated appear to be too low for fracture initiation."

Studies indicate that the highest risk of sudden pipe rupture in the primary system arises from the failure of supports of a major component. We can see no reason to be sanguine about the safety of operating nuclear power plants with the largest, heaviest component in the primary system supported on a structure, parts of which are fully brittle. This is unsafe by any type of analysis. The average stress may be compressive, but it isn't the average stress which would determine the failure of the structure. These supports are welded structures so there are regions with tensile stresses as high as the yield stress. They operate in a temperature gradient so there will be thermal stresses which are tensile in the cold (less ductile) regions. They are uninspected so we have no real idea of what kinds of flaws are present, and flaw size is critical in any meaningful failure analysis.

It would be imprudent to operate nuclear power plants with brittle structures supporting the pressure vessels. We recommend that an early effort be made to gain answers to the following questions:

1) Is the temperature of the support structure of the reactor pressure vessel in any operating plant now below its nil ductility transition temperature (NDTT)?

2) Will the temperature of the support structure of the reactor pressure vessel in any operating plant drop below its NDTT before the plant's license expires?

We hope and suspect that the answer to the first question is "no." However, it is not clear that we know this with any certainty. The research program mentioned in your memorandum is necessary and desirable, but it is not clear that it will answer the safety-related questions noted above in a timely manner.

Sincerely,

W Kerr

William Kerr
Chairman

Reference:
Memorandum from Victor Stello, Jr., EDO, to William Kerr, ACRS, dated October, 7, 1987, Subject: ACRS Comments on the Embrittlement of Structural Steel

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

August 12, 1987

The Honorable Lando W. Zech, Jr.
Chairman
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Zech:

SUBJECT: ACRS COMMENTS ON THE ADVANCE NOTICE OF PROPOSED RULEMAKING: DEGREE REQUIREMENTS FOR SENIOR OPERATORS

During the 328th meeting of the ACRS, August 6-8, 1987, and our 327th meeting, July 9-11, 1987, we discussed SECY-87-101, "Issues and Proposed Options Concerning Degree Requirements for Senior Operators," which was prepared in response to public comments on the proposed rule. Meetings of our Subcommittee on Human Factors were also held on July 15, 1986 and June 24, 1987 to discuss this issue with the NRC Staff. During these meetings, we had the benefit of presentations by the NRC Staff as well as representatives of the Westinghouse Electric, KMC, and Delian corporations. We also had the benefit of the documents referenced.

On May 31, 1986 the NRC published an Advance Notice of Proposed Rulemaking (ANPRM) to require all applicants for a Senior Reactor Operator (SRO) license to possess a baccalaureate degree in engineering or physical science after January 1, 1991. Two hundred letters of public comment were received in response to the ANPRM of which approximately 98% indicated opposition to the NRC's proposal.

The nuclear utility industry and the NRC have endorsed a systems approach to performance based training. At the heart of performance based training is a detailed Job and Task Analysis (JTA) which analyzes the many tasks that must be performed to carry out the various jobs of personnel filling positions in nuclear power plants, including the position of SRO. The tasks are further analyzed to determine the various knowledges, skills, and abilities (KSAs) that one must possess to perform the tasks. The analysis continues further to determine whether the KSAs should be obtained through formal education or through specific training in the classroom, in the laboratory, at a simulator, or by self-study.

A number of JTAs have been performed by licensees as part of the conversion to performance based training; analysis of these JTAs has not shown that a college degree is necessary for Senior Reactor Operators to perform the tasks of their jobs to ensure safety of plant operations. A Peer Advisory Panel appointed by the Commission came to the same conclusion in 1982 and recommended against a degree requirement for SROs. We have not been informed of any technical rationale for requiring a degree for SROs at nuclear power plants; we conclude, therefore, that a degree requirement for all SROs is primarily a policy issue.

We strongly support the concept of having engineering expertise on each shift. The Commission's requirement of a Shift Technical Advisor (STA) was a step in that direction. Further, the Commission's provision of the option to combine the STA function with one of the SRO positions was a step to encourage greater integration of the resulting engineering expertise into shift operations. The Committee endorsed both of these actions. The NRC Staff indicates that the percentage of SROs with a baccalaureate degree in engineering or physical science has increased from 1[illegible] in 1980 to 28% in 1987.

We are informed that the primary reasons for considering requiring all SROs in the future to have degrees is to enhance professionalism in reactor operations and to make it more likely that the higher management positions in nuclear utilities will be filled by individuals with plant operations experience. We endorse these purported goals but question whether they will be realized through the proposed indirect approach of requiring degrees of all SROs. We believe there is a more direct approach to achieving these goals than through the proposed rulemaking.

We recommend that the Commission formulate more specifically its concerns and the goals it desires to achieve. The Commission then should meet with appropriate licensee representatives (e.g., NUMARC) to convey the need for increased attention to the areas of concern. The NRC Staff and the licensees should then work to develop solutions, programs, and schedules for implementation of any changes from current practice deemed necessary. We realize that proposed rulemaking is one method to generate sufficient attention to encourage licensee initiative; however, we believe a more direct and less adversarial approach is preferable when the proposed action is not driven by clearly identified public safety concerns.

In summary, although the purported goals of the proposed rulemaking are laudable, we think that the depth of the concern about adverse effects of the proposed rule should be reconsidered; many of the comments were received from individuals who are knowledgeable about personnel considerations in the work place. We recommend a more direct approach to identifying and addressing the Commission's concerns.

Additional comments by ACRS member Glenn A. Reed are presented below.

Sincerely,

W Kerr

William Kerr
Chairman

Additional Comments by ACRS Member Glenn A. Reed

I applaud the ACRS letter and wish to add further support to it. As a person who earned a university engineering degree and one who held an NRC SRO license, I am opposed to the degree requirement for SROs, as in my opinion it is not needed from a job task analysis viewpoint, is not in the interest of licensed personnel morale, is not needed in the interest of best safety of operations, and would lessen the experience qualifications of SRO personnel. I have found that a college degree in engineering or applicable science will probably ensure that an SRO candidate will have an acceptable enough intelligence quotient to be able to take on-site training. However, there is no assurance from the college degree achievement that the SRO candidate will have the even more important qualifications of mechanical comprehension, logical reasoning, and appropriate personality.

My thirty plus years of hiring and working with licensed operators has convinced me that acceptable performance in a battery of aptitude tests (IQ, mechanical comprehension, logical reasoning, and personality traits), coupled with appropriate experience and training, will provide the best SRO performers and people in overall shift charge. My experience also has convinced me that the Shift Technical Advisor concept that was endorsed some years ago by the NRC can provide the best engineering support, and the best future promotional cross-fertilization into utility top management, and into the vendor design field.

References:
1. SECY-87-101, April 16, 1987, Issues and Proposed Options Concerning Degree Requirements for Senior Operators.

2. Federal Register, Vol. 51, No. 104, Page 19561, Friday, May 30, 1987, Advance Notice of Proposed Rulemaking, 10 CFR Parts 50 and 55, Degree Requirements for Senior Operators at Nuclear Power Plants.

3. Comments pertaining to the Advance Notice of Proposed Rulemaking - Degree Requirements for Senior Operators, KMC, Inc., September 29, 1986.

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

June 10, 1987

The Honorable Lando W. Zech, Jr.
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Zech:

SUBJECT: PROPOSED INTERNATIONAL WORKSHOP ON QUALITY IN DESIGN AND CONSTRUCTION OF NUCLEAR POWER PLANTS

We believe that the present hiatus in licensing actions and the near completion of construction of the present generation of nuclear power plants provides an excellent opportunity for the NRC to reexamine the question of how best to achieve quality in design, manufacture, and construction.

Because of the great reliance placed by the NRC Staff on extensive and expensive formal quality assurance (QA) programs, it is important that we attempt to determine whether these QA programs deserve credit for actually achieving quality in the constructed plant. It now is assumed that programmatic or implementation deficiencies in the QA program are indicative of corresponding deficiencies in the quality of the plant. Conversely, it is assumed that if there are no deficiencies found in the QA program, the plant quality will be very high. Unfortunately, we have been unable to determine that either of these assumptions is valid.

Because other countries with substantial nuclear power programs seem to have achieved levels of quality even higher than we have in the United States, and because most of those countries do not require or utilize the kind of quality assurance programs that we do, it would be informative to review and discuss with them their philosophies, procedures, and practices to achieve quality in their plants.

We recommend, therefore, that you consider the organization and sponsorship of an International Workshop on Quality in Design and Construction. We would be happy to provide further thoughts on the agenda for such a workshop and would be willing to participate in the planning to the extent that our resources permit.

Sincerely,

W Kerr

William Kerr
Chairman

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

September 15, 1987

The Honorable Lando W. Zech, Jr.
Chairman
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Chairman Zech:

SUBJECT: ACRS COMMENTS REGARDING PROPOSED INTERNATIONAL WORKSHOP ON QUALITY IN DESIGN AND CONSTRUCTION OF NUCLEAR POWER PLANTS

In our June 10, 1987 letter to you we proposed an international workshop on quality in design and construction of nuclear power plants.

In a memorandum dated August 7, 1987 (attached), Mr. Stello agreed with the importance of assessing the contribution of quality assurance (QA) programs in assuring quality in nuclear power plants, but concluded that consideration of the proposed workshop should be deferred because the construction of new nuclear power plants in the United States is nearing completion and because the NRC Staff is concentrating on improving nuclear power plant operations.

The fact that there is a pause in the design and construction of new plants in the United States is one of the reasons we think that now is the time to evaluate the worth of existing regulations and associated regulatory guidance in attaining quality. The time for evaluation and possible change of the regulations and associated guidance is not after new applications have been submitted in accordance with existing regulations. We believe that the present hiatus in licensing actions and the near completion of construction of the present generation of nuclear power plants provides an excellent opportunity for the NRC to reexamine in an orderly fashion the question of how best to achieve quality in future plants. Further, we believe that such reexamination should take place before the agency's and the industry's memory of past difficulties with current regulations and guidance has been blurred or lost.

Although the part played by quality assurance in operation of power plants and in licensing of waste repositories was not made explicit in our letter of June 10, 1987, it was not our intent to exclude these important questions. In fact, we believe that the proposed international workshop on quality should specifically include discussion of these items. We believe that there is much that could be gained from better understanding of how quality in plant operations is achieved in the United States and abroad.

We believe that our recommendation to the Commission to reexamine at this time the question of how to achieve quality is consistent with the draft NRC Strategic Plan which specifically makes the assumption that quality problems can be expected in the future. Further, we believe that the recommendation is consistent with the goal of the NRC which is stated in the draft strategic plan to "ensure that nuclear power plants under construction are designed and constructed properly and are ready for safe operation," as well as being consistent with the stated goal to "prepare for future reactor licensing activities."

We would be pleased to meet and discuss this matter in greater detail with you in the near future.

Sincerely,

WKerr

William Kerr
Chairman

Attachment:
Memorandum from Victor Stello, Jr., EDO, to William Kerr, ACRS, dated August 7, 1987, "Proposed International Workshop on Quality in Design and Construction of Nuclear Power Plants"

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

August 7, 1987

MEMORANDUM FOR: William Kerr, Chairman
Advisory Committee on Reactor Safeguards

FROM: Victor Stello, Jr.
Executive Director for Operations

SUBJECT: PROPOSED INTERNATIONAL WORKSHOP ON QUALITY IN AND CONSTRUCTION OF NUCLEAR POWER PLANTS

The purpose of this memorandum is to respond to your June 10, 1987, letter to Chairman Zech concerning a proposed international workshop on quality in design and construction.

I agree with the importance of assessing the contribution of the licensees' quality assurance (QA) programs in assuring quality in nuclear power plant design and construction. In this regard, the staff has implemented several key initiatives: readiness reviews, integrated design inspections (IDIs), and construction appraisal team inspections (CATs).

As current construction of nuclear power plants in the United States nears completion, we are focusing primary staff attention on improving nuclear power plant operations by applying available staff resources to initiatives that include performance-oriented QA inspections in lieu of the programmatic-type QA inspections, safety system functional inspections (SSFIs), and safety systems outage modification inspections (SSOMIs). We are also continuing to ensure high quality design, engineering and construction in support of modifications to existing operating plants. These types of efforts should improve our efforts to foster programs that provide a better assurance of quality for standardized plants.

Because of the dedication of the NRC staff to these initiatives, I feel that we should defer consideration of the proposed workshop at this time. I believe it would be more appropriate for the Commission to reconsider such a proposal when a clearer picture emerges of the next generation of nuclear power plants in the United States.

Victor Stello, Jr.
Victor Stello, Jr.
Executive Director
for Operations

CONTACT: Jack W. Roe, NRR
49-24803

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

July 15, 1987

Mr. Victor Stello, Jr.
Executive Director for
Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Stello:

SUBJECT: ACRS COMMENTS ON INTERNATIONAL COOPERATION ON RESEARCH RELATED TO RADIATION PROTECTION

During the 327th meeting of the ACRS, July 9-11, 1987, it was brought to our attention that members of the NRC Office of Nuclear Regulatory Research are attempting to develop a system for the coordination of research being conducted in various countries on the biological effects and control of ionizing radiation.

These efforts hold promise for assuring that key problems are effectively addressed and for reducing unnecessary duplication and the wasting of resources. We endorse these efforts and encourage their support.

Sincerely,

W Kerr

William Kerr
Chairman

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

November 10, 1987

The Honorable Lando W. Zech, Jr.
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Zech:

SUBJECT: ACRS COMMENTS ON THE DEVELOPMENT OF RADIATION PROTECTION STANDARDS

During the 331st meeting of the Advisory Committee on Reactor Safeguards, November 5-7, 1987, we met with Floyd L. Galpin, Office of Radiation Programs, U.S. Environmental Protection Agency (EPA), and Robert E. Alexander, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission (NRC), to discuss current developments related to radiation protection standards. These discussions included reports on the efforts of (1) EPA to establish individual dose rates for members of the public that would be considered to be "below regulatory concern" (BRC), and (2) an interagency committee, coordinated by EPA with NRC support, that is engaged in developing guidance for federal agencies on radiation protection of the public. These topics were also subjects of discussion by our Waste Management Subcommittee during its meeting on October 15-16, 1987.

Current EPA efforts are being directed primarily to developing limits on dose rates from low-level radioactive wastes, including the development of dose rates that are BRC, for members of the public. Several proposals on this topic from outside organizations have been reviewed and endorsed by the EPA's Science Advisory Board. As such, this work holds promise for alleviating some of the problems being encountered in the management and disposal of such wastes.

Although these efforts have revealed inconsistencies in existing radiation protection standards (which will require considerable efforts to resolve), and although problems remain (such as clarifying distinctions in dose rates considered to be BRC and those considered to be <u>de minimis</u>), we are very encouraged by these activities. They hold <u>promise</u>, not only of providing a coherent system of radiation protection standards, but also of placing the risks from low radiation dose rates in better perspective.

For these reasons, we recommend that the NRC continue its support of and lend encouragement to the work of the interagency committee and the related efforts of the EPA.

Sincerely,

W Kerr

William Kerr
Chairman

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

January 15, 1987

The Honorable Lando W. Zech, Jr.
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Chairman Zech:

SUBJECT: ACRS RECOMMENDATIONS ON IMPROVED SAFETY FOR FUTURE LIGHT WATER REACTOR PLANT DESIGN

During the 321st meeting of the ACRS, January 8-10, 1987, we completed our discussion of improved safety requirements and objectives for future light water reactor power plants (LWRs). This discussion began during the 316th ACRS meeting, August 7-9, 1986. The scope of our present comments is limited to nuclear power plant design. Other factors, such as plant operation and management, are necessarily involved, but are beyond the scope of our present remarks.

The ACRS has on several previous occasions recommended that future LWRs should be designed to be safer than current LWRs. This is not to ignore the excellent safety record thus far of LWRs in the United States. We believe this increased safety can be achieved with reasonable economy because better technology is available today. Improved concepts for plants and improved understanding of risks have been developed over a generation of experience in design, operation, and analysis. But, not all of these concepts have been incorporated into the newest reported LWR designs. We believe many of these concepts can be incorporated with acceptable effect on plant cost or operating efficiency. With the expectation that future plants will be standardized, the next group of plants to be licensed will probably set the safety design philosophy, and even details of implementation, to be used in nuclear power plants for several decades.

The mean estimates of risk from generation of electricity by the use of nuclear energy are at least as low as those for generation by other methods. However, the acceptability of these estimates is much affected by the large uncertainty associated with them. A compelling reason for implementing improvements -- apart from the fact that improvements are possible -- is to reduce the uncertainty in the risk estimates.

Future plants should be able to survive a wider spectrum of off-normal challenges and mistreatments. For example, normal operating systems should be forgiving of most operational errors and imperfections in maintenance. Accident management and mitigation systems should be designed, not for a narrow set of design-basis accidents, but to reasonably accommodate a broad range, variety, and time sequence of threats.

Our recommendations are based on insights provided from quantitative risk analyses, lessons learned from operating experience, and continuing concerns. In the sections that follow, we list and discuss a number of possible safety improvements. Several of these overlap, and we do not expect that all of them should be implemented. Rather, we offer them with the belief that each is worthy of serious consideration in connection with future designs.

1. Dedicated and Protected Decay Heat Removal System (DHRS)

   We recommend for consideration that future LWRs include a dedicated, protected, redundant, decay heat removal system having its own power, fuel, and water supply, with a capability for makeup, including coolant lost from very small LOCAs, and for recirculation from the containment sump. This system should have a large seismic capability such that its function is not threatened by earthquakes having an occurrence likelihood in the range 10^{-5} to 10^{-6} per year. There should be similar protection and seismic capability for the primary system and all components whose specific function is required for proper operation of the dedicated decay heat removal system, as well as protection against fires, flooding, and adverse environmental effects. This system should be capable of actuation but not termination from the control room.

   We list this item first because the provision of such a system would alleviate our concerns in several areas, including the following:

   - If the DHRS is protected against fire, internal or external floods, sabotage by an insider or by terrorists, and earthquakes at the 10^{-5} to 10^{-6} probability level, the degree of protection required of other portions of the plant against such events could be relaxed in many instances. In addition to the economies these reductions might lead to, we believe that they might lead to relaxation or removal of many of the impediments to access and flexibility of operation that are now imposed by security and fire control.

   - The loss of all sources of AC power, both off-site and on-site (station blackout), would be of less concern if a DHRS is provided. However, vital DC power and certain vital cooling functions (such as cooling of primary pump seals in a PWR) now performed by using AC power would have to be dealt with appropriately.

   In some of the further recommendations that follow, we indicate that the identified needs would be reduced, or perhaps eliminated, if a dedicated, protected, decay heat removal system were provided.

2. Safety Train Redundancy

The general principle of "N+2" trains should be adopted for active, safety-related functions. N is defined as the number of trains required to perform a necessary safety function. N is equal to one if the train has 100% capacity to perform the function. N is equal to two if each train has 50% capacity. Thus, an "N+2 rule" would require three 100% trains, or four 50% trains. Each of the N trains would have its own independent support systems. Each train would be physically separate from the others, and diverse designs or equipment should be considered if this can be shown to provide a significant safety advantage. Exceptions to this general principle should be permitted for systems providing functions with low risk potential and for systems which can be demonstrated to be exceptionally robust and reliable.

The proposed high level of functional capacity could be used to improve plant availability by use of Technical Specifications which permit one of the extra trains to be out of service for maintenance and testing for somewhat longer periods than is now the practice for the first train of redundancy.

3. Design of Containment Systems

The need to mitigate the consequences of certain severe accidents should be considered explicitly in the design of containment systems (structures, penetrations, sprays, vents, etc.). The severe accident sequences to be considered should be those for which the mitigation provided by the containment systems is required to meet the Commission's proposed general performance guideline that the overall mean frequency of a large release of radioactive materials to the environment from a nuclear power plant accident should normally be less than 1 in 1,000,000 per reactor year of operation. The severe accident sequences that need not be considered are those of sufficiently low probability that the releases, unmitigated by specially designed containment systems, will in the aggregate not exceed this objective.

4. Protection Against Sabotage

We are not of one mind on the issue of the extent to which LWRs should be protected against the threat of damaging sabotage by terrorists and insiders.

On the one hand, there is reason to believe that certain design choices can lead to inherently better resistance against such a threat, even if these choices are not specifically directed against sabotage. For example, control rooms can be positioned so they are away from the exterior ground level and protected from truck bombs by existing massive concrete structures. Good physical separation of redundant safety trains may provide significant inherent protection. Some of us favor hardening, or separation, or other protection of most vital functions such that they are relatively well protected against transportable explosives. If included in the original design, part of these changes should result in modest added cost or modest loss of other beneficial plant characteristics.

On the other hand, some of the members are not convinced there is reason to believe nuclear power plants are particularly attractive targets for saboteurs. If a terrorist aims to actually cause injury to large numbers of the public, there are far easier and more effective targets throughout the country. Also, with 120 operating plants [today's population] built to a lower level of sabotage protection and a new set of plants built to a higher level of sabotage protection, this discrepancy will surely be noted and taken into account by a terrorist in the selection of a specific target, if the aim is to cause physical harm to the public. It appears to these members that the resources society allocates for defense against terrorism would be more effectively used in areas other than nuclear power.

In the case of the insider, the ACRS believes the threat is of low probability. This should not, however, discourage prudent design features which could impede insider actions or reduce the likelihood of success.

5. Fire Protection

Those responsible for conducting probabilistic risk assessments (PRAs) have not been very successful in quantifying the risk from large fires involving essential reactor systems. As a result, the real benefit of existing fire protection provisions and backfits remains uncertain. We believe future LWRs should be designed so that cold shutdown of the plant using safety-grade equipment can be accomplished quickly (within 24 hours) in the event of any single fire which may burn up to 3 hours. Physical separation and protective barrier or compartment arrangements should include a reasonable accounting for the adverse effects of the spread of heat and the products of combustion beyond the fire zone, including consequential spurious actuation of fire mitigation features and the resulting damage to safe shutdown equipment. Fire mitigation features should be designed to function properly, and not to spuriously actuate, during or after a seismic event.

If the plant has a DHRS as discussed above, only those other portions of the plant vital to accomplishing safe shutdown would need to be protected against fire consistent with the more stringent requirements listed above.

6. Anticipated Transients Without Scram (ATWS)

We suggest that design features be introduced that would make an ATWS event a much less serious, if not a negligibly small contributor, to risk. For PWRs this might involve some combination of increased negative moderator temperature coefficient of reactivity and increased pressure-relieving capability for the primary system. For BWRs a partial contribution would be made by something approaching 100% relief capability in the event of turbine trip or main steam isolation valve closure. We also suggest that the combination of control and safety systems be examined for reliability, as well as for testing and maintenance of the systems, to reduce the need for some of what are now considered to be safety-related scrams, as well as to reduce the number of spurious scrams.

7. Systems Interactions

Operating experience and reviews of existing nuclear power plants have provided evidence of unanticipated adverse interactions from supposedly separate systems. These supposedly separate systems sometimes interact in unanticipated ways because they are dependent on common support systems (such as power supplies, common piping systems, etc.) or because they share the same or adjoining physical space. Those people responsible for performing PRAs can successfully incorporate the effect of these interactions only if they are known and understood and if probabilities of occurrence can be established. We believe that further effort is warranted to develop techniques and processes which can seek out and eliminate such interactions.

8. Electric Power Systems

We believe that the frequency of transients and spurious reactor scrams should be reduced by providing electric power supplies that are less vulnerable to transmission network disturbances. We recommend that General Design Criterion 17 be revised to require that the circuit which is provided to be immediately available to cope with a LOCA be the normal power supply to the plant auxiliaries and safety systems and be supplied continuously and unswitched from the low side of the main stepup transformer during and throughout startup, operation, and shutdown of the nuclear generating unit.

We believe that the capability of a plant to cope with the loss of all off-site power can be improved. For one thing, the proposed resolution of Unresolved Safety Issue A-44, Station Blackout, should be implemented in the design of future plants.

For another, the reliability of on-site AC power sources can be enhanced by designing the nuclear system with sufficient steam bypass, feedwater inventory and make-up, and run back capability to sustain unit load rejection from 100% power and to run back to "house" electrical load, or by providing an additional, preferably diverse, standby electrical generating unit. The need for these features would be reduced if a DHRS is provided, as discussed in Item 1 above.

9. Probabilistic Seismic Design

   Important safety systems should be explicitly designed using probabilistic seismic design methodology to survive and function during and after severe seismic events. Only survivability and those functions needed to bring to and hold the reactor at cold shutdown need be considered. A DHRS such as discussed above would reduce the number of structures and systems requiring very stringent seismic design.

10. Primary Pressure Boundary

   We recommend that the primary system pressure boundary be designed and fabricated to minimize the number of welds and optimize the ease of inspecting them.

11. Dedicated Systems and Sharing

   There should be minimum sharing of equipment, flow paths, and support facilities among nominally separate systems.

12. Control Room Protection for Severe Accidents

   Safe habitation of the control room and other necessary facilities should be ensured in the event of an accident that results in a large release of radioactive materials outside containment. For multi-unit sites, this requirement applies to both the damaged unit and other units on the site.

Additional comments by ACRS Members H. Lewis, F. Remick, P. Shewmon, and D. Ward are presented below.

Sincerely,

WKerr

William Kerr
Chairman

Additional comments by ACRS Members H. Lewis, F. Remick, P. Shewmon, and D. Ward.

This is a camel of a letter, describing a camel of a reactor. We have no reason to doubt that each of the features recommended in the letter may improve safety, nor do we have any reason to believe that there are not better and more cost-effective alternatives. This problem is compounded to the extreme by putting them all together.

The purpose of this letter is presumably to distill the Committee's observations and experience with the current generation of reactors, designed over the last few decades, and to put that experience to work in expressing a design philosophy for the next generation of reactors. There is no hint of a philosophy, but instead a laundry list of improvements, all unanalyzed. Though the Committee has often recommended that the next generation be safer than the past, that recommendation has never been justified. It may be right, but seems to be inconsistent with the Commission's Safety Goal Policy. There is no doubt in our minds that, with new technology and years of experience, a new generation can be either safer at comparable cost and level of complication, or equally safe at lower cost and greater simplicity, and that choice is so fundamental that it is, in our view, not responsible for the Committee to opt for greater safety and greater complication without analysis or justification.

We believe one can learn from experience and that the next generation must inevitably be better than the past (and thereby safer), but we are uncomfortable about designing those reactors in committee.

Additional comments by ACRS Member, David A. Ward.

I disagree with the Committee's recommendation that future LWRs should include a dedicated, bunkered decay heat removal system. In my opinion, the safety advantage from such a system is highly uncertain and likely to be very slight or even negative. The cost would be great and there would be added complexity in operations. I believe added reliability offered by adoption of the N+2 principle with some diversity and separation of trains is adequate and preferable.

The promises of trade offs, e.g., relaxation of requirements on mainline systems, are phantoms. A systematic study to determine what should be included in a bunkered system and whether there would indeed be important trade offs might be warranted at this time, but the Committee has not made such a study. The recommendation is a hip shot.

The Committee has elected not to make recommendations relative to either of a pair of weaknesses in LWRs which I believe make them the object of criticism from the proponents of new reactor concepts. These are: 1) absence of a backup scram system and 2) the fact that every scram, real or spurious, becomes a challenge to the plant because of the necessity for emergency feedwater. I believe consideration should be given to development of an independent backup scram system for LWRs. This would include not only independent sensor and control logic, but also an additional system of absorber rods or other material (possibly a liquid) to rapidly and reliably enter the core. Further, I believe there should be consideration of a passive or continuously operating decay heat removal system so that a reactor scram will not be a challenge, but instead always be an unambiguous shift to a safer operating mode.

Beyond these two specific points, I believe the best approach for the NRC to take in implementing safety improvements, such as those suggested in this letter, in LWRs of the future is to incorporate them into a revised set of General Design Criteria. Although iteration with designers and licensees will be necessary, the improvement process will best be served by establishing a clear new basis at the beginning.

In addition, I am concerned that the concept of quality assurance, as applied in the nuclear power industry, has not been successful. I do not, of course, question the need for quality nor do I have major concerns about the quality of existing plants. However, I do question whether QA has had much to do with either. This might not be so troublesome except that QA as practiced is very expensive and uses resources that might better be spent in other activities, including more effective reactor safety programs. I suggest that the present hiatus in plant design and construction provides an opportunity for the Commission to rethink its commitment to the present concepts and practices of QA.

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

July 15, 1987

The Honorable Lando W. Zech, Jr.
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Zech:

SUBJECT: ACRS COMMENTS ON IMPROVED SAFETY FOR FUTURE LIGHT WATER REACTORS

During the 327th meeting of the ACRS, July 9-11, 1987, we discussed your request (see Reference) for the name of an existing plant that incorporates the desirable features recommended for consideration in our letter to you dated January 15, 1987 on Improved Safety for Future Light Water Reactor Plant Design. To the best of our knowledge, we believe that most of those features are incorporated in the Federal Republic of Germany KWU-Standard PWR plant designed, licensed and constructed under the KONVOI process. Examples are Isar 2, Emsland, and Neckarwestheim 2, which are near completion.

Additional comments by ACRS Member Harold W. Lewis are presented below.

Sincerely,

W Kerr

William Kerr
Chairman

Additional Comments by ACRS Member Harold W. Lewis

I am greatly concerned lest the Commission may have misunderstood the list in our earlier letter as a list of recommendations of features the Committee views as desirable for incorporation into future reactors. It should be emphasized that those were recommended for consideration only. None have been sufficiently analyzed by the Committee to justify a stronger interpretation.

Reference:

Memorandum dated April 22, 1987 to Raymond F. Fraley, ACRS, from John C. Hoyle, Assistant Secretary, Subject: Staff Requirements - Periodic Meeting with Advisory Committee on Reactor Safeguards

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

September 15, 1987

The Honorable Lando W. Zech, Jr.
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Zech:

SUBJECT: ACRS COMMENTS ON IMPROVED SAFETY FOR FUTURE LIGHT WATER REACTORS

During the 329th meeting of the ACRS, September 10-12, 1987, we discussed two requests transmitted in the memorandum from John C. Hoyle, Assistant Secretary, NRC, to Raymond F. Fraley, Executive Director, ACRS, dated April 22, 1987 (see Reference). The ACRS Subcommittee on Future Light Water Reactor Designs had previously discussed these requests during a meeting on September 9, 1987.

The first request was that "the ACRS pursue its review of the experience and design features of some of the European plants." We intend to continue such a review and will keep the Commission informed of our findings as appropriate.

The second request was that the ACRS "address the feasibility, benefit, and cost effectiveness of selected and combined systems recommended in the Kerr to Chairman Zech letter dated January 15, 1987. The review should include plant reliability, challenges, complexity, and burden on plant and maintenance personnel." We believe that such a study clearly is desirable. However, it would require consideration of many aspects of design other than safety and is beyond our capabilities and resources. For these reasons, it is more appropriate as a task for the NRC Staff or a contractor.

We would be pleased to discuss this with you further.

Additional comments by ACRS Member Glenn A. Reed are presented on the following page.

Sincerely,

WKerr

William Kerr
Chairman

Additional Comments by ACRS Member Glenn A. Reed

As you know, both the General Electric Company and Westinghouse Electric Corporation have stated that their advanced LWR designs (on the drawing boards) do indeed incorporate most or all of the features mentioned in the ACRS letter of January 15, 1987. As should be realized, there's a long path between the drawing board and a built operating reactor, and therefore I recommend that the NRC sponsor an in-depth study as a follow-on to USI TAP A-45 that addresses the most important recommendation of the ACRS January 15, 1987 letter, the recommendation on a dedicated decay heat removal system. The follow-on study should address decay heat removal for future LWRs and the systems, diversity of systems, redundancy of components, and the other complex safety influencing aspects such as security and fire. The operating reactor KONVOI should not be excluded from the study. It is my opinion that an in-depth study may reveal cost savings and improved operating and emergency potential for future LWRs. In particular, I feel that the use of a backup primary blowdown (dedicated) depressurization and decay heat removal system for PWRs will provide improved operations, less operating burden, fewer security demands, and reduced core melt probability.

Reference:

Memorandum dated April 22, 1987 to Raymond F. Fraley, ACRS, from John C. Hoyle, Assistant Secretary, Subject: Staff Requirements - Periodic Meeting with Advisory Committee on Reactor Safeguards

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

April 13, 1987

Mr. Victor Stello, Jr.
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Stello:

SUBJECT: ACRS ACTION ON THE PROPOSED REGULATORY GUIDE EE 404-4, "ENVIRONMENTAL QUALIFICATION OF CONNECTION ASSEMBLIES FOR NUCLEAR POWER PLANTS"

During our 324th meeting, April 9-11, 1987, the members of the Advisory Committee on Reactor Safeguards discussed a report from our Subcommittee on Regulatory Activities regarding the proposed Regulatory Guide EE 404-4, "Environmental Qualification of Connection Assemblies for Nuclear Power Plants."

As a result of this discussion, we concur in the NRC Staff's proposal to issue the subject Guide for public comment. After the public comment period, we expect to review the proposed final version of this Guide together with the public comments and the NRC Staff's response to them.

Sincerely,

WKerr

William Kerr
Chairman

Reference:

Proposed Regulatory Guide EE 404-4, Draft 2, "Environmental Qualification of Connection Assemblies for Nuclear Power Plants," dated September 8, 1986, transmitted to the ACRS by a memorandum from Guy A. Arlotto to Raymond F. Fraley, dated March 24, 1987.

cc: S. J. Chilk, SECY
T. Rehm, EDO
E. Beckjord, RES
G. A. Arlotto, RES
S. K. Aggarwal, RES
C. Bartlett, RES
R. Hernan, NRR

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

June 9, 1987

Mr. Victor Stello, Jr.
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Stello:

SUBJECT: ACRS ACTION ON PROPOSED REVISION 2 OF REGULATORY GUIDE 1.100, "SEISMIC QUALIFICATION OF ELECTRIC AND MECHANICAL EQUIPMENT FOR NUCLEAR POWER PLANTS"

During our 326th meeting, June 4-6, 1987, the members of the Advisory Committee on Reactor Safeguards discussed a report from our Subcommittee on Reliability Assurance regarding proposed Revision 2 of Regulatory Guide 1.100, "Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants."

As a result of this discussion, we concur in the NRC Staff's proposal to issue the subject Guide for public comment. After the public comment period, we expect to review the proposed final version of this Guide together with the public comments and the NRC Staff's response to them.

Sincerely,

WKerr

William Kerr
Chairman

Reference:
Memorandum from Guy A. Arlotto, Office of Nuclear Regulatory Research to Raymond F. Fraley, ACRS, dated May 8, 1987, transmitting:

1. Proposed Revision 2 to Regulatory Guide 1.100, Draft 4, "Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants," dated December 4, 1986, and

2. Proposed IEEE Standard 344-1987, "Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations," Draft 9, dated July 1986

cc: S. J. Chilk, SECY
T. Rehm, EDO
E. Beckjord, RES
G. A. Arlotto, RES
S. K. Aggarwal, RES
C. Bartlett, RES
R. Hernan, NRR

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

August 11, 1987

Mr. Victor Stello, Jr.
Executive Director for Operations
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Stello:

SUBJECT: ACRS ACTION ON PROPOSED REGULATORY GUIDE EE 006-5, "QUALIFICATION OF SAFETY-RELATED LEAD STORAGE BATTERIES FOR NUCLEAR POWER PLANTS"

During the 328th meeting of the ACRS, August 6-8, 1987, we discussed a report from our Subcommittee on Reliability Assurance regarding proposed Regulatory Guide EE 006-5, "Qualification of Safety-Related Lead Storage Batteries for Nuclear Power Plants," Draft #2.

As a result of this discussion, we concur in the NRC Staff's proposal to issue the subject Guide for public comment. After the public comment period, we expect to review the proposed final version of this Guide together with the public comments and the NRC Staff's response to them.

Sincerely,

W Kerr

William Kerr
Chairman

References:
Memorandum from Guy A. Arlotto, Office of Nuclear Regulatory Research to Raymond F. Fraley, ACRS, dated June 12, 1987, transmitting:

1. Proposed Regulatory Guide EE 006-5, "Qualification of Safety-Related Lead Storage Batteries For Nuclear Power Plants," Draft #2

2. IEEE Standard 535-1986, "IEEE Standard for Qualification of Class 1E Lead Storage Batteries for Nuclear Power Plants," June 25, 1986

cc: S. J. Chilk, SECY
T. Rehm, EDO
E. Beckjord, RES
G. A. Arlotto, RES
S. K. Aggarwal, RES
C. Bartlett, RES
R. Hernan, NRR

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

October 14, 1987

Mr. Victor Stello, Jr.
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Stello:

SUBJECT: ACRS ACTION ON PROPOSED FINAL REGULATORY GUIDE (TASK EE 404-4), "ENVIRONMENTAL QUALIFICATION OF CONNECTION ASSEMBLIES FOR NUCLEAR POWER PLANTS"

During the 330th meeting of the Advisory Committee on Reactor Safeguards, October 8-10, 1987, we concurred in the regulatory position proposed in Regulatory Guide (Task EE 404-4), "Environmental Qualification of Connection Assemblies for Nuclear Power Plants."

Mr. C. J. Wylie did not participate in the Committee's deliberations regarding this matter.

Sincerely,

WKerr

William Kerr
Chairman

Reference:
Proposed Final Regulatory Guide (Task EE 404-4) (May 1987 version) transmitted by memorandum dated September 21, 1987 from G. A. Arlotto, Office of Nuclear Regulatory Research, to R. F. Fraley, ACRS, Subject: Regulatory Guide EE 404-4, "Environmental Qualification of Connection Assemblies For Nuclear Power Plants."

cc:
S. J. Chilk, SECY
T. A. Rehm, EDO
E. Beckjord, RES
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

May 13, 1987

The Honorable Lando W. Zech, Jr.
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Zech:

SUBJECT: ACRS COMMENTS ON AN IMPLEMENTATION PLAN FOR THE SAFETY GOAL POLICY

During the 325th meeting of the Advisory Committee on Reactor Safeguards on May 7-9, 1987, we formulated comments on the NRC Staff's proposed implementation plan for the Commission's August 4, 1986 Policy Statement on Safety Goals for the Operations of Nuclear Power Plants. This topic had previously been considered during our 321st, 322nd, 323rd, and 324th meetings in January, February, March, and April of 1987, and during a subcommittee meeting on January 7, 1987. In our review we had the benefit of discussions with the NRC Staff and the documents listed.

We do not consider the current Staff proposal (Reference 1) suitable as a plan for implementing the Safety Goal Policy. Instead, we propose a plan of three elements:

1. Use of safety goal criteria by the NRC Staff to judge the adequacy of regulation rather than to make regulatory judgments about specific plants.

2. Recognition and formulation of an explicit hierarchical structure among the interrelated criteria in the overall goal.

3. Continuation of a program to make risk estimates for specific plants, as a sampling process to assist in the evaluation of regulation.

USE OF THE SAFETY GOAL

It appears that the plan proposed by the NRC Staff is intended for their use in judging whether a specific nuclear power plant can be permitted to operate or continue to operate. If the Staff concludes that a plant does not meet certain quantitatively stated elements of the Safety Goal Policy Statement, an evaluation would be made of possible changes, for example in design, equipment, or procedures, to determine if such changes would result in an improved or acceptable level of risk in a cost-effective manner.

We do not believe that probabilistic risk assessment (PRA), on which the proposed process is based, is sufficiently developed to be used to make narrowly differentiated decisions about specific plants. Rather, the Safety Goals should be used in a more global manner to judge the suitability of existing regulations and regulatory practices, or to assist in formulating whatever changes are necessary to provide confidence that nuclear plants are operating within an envelope established by the Safety Goals.

We continue to believe that systematic examinations of individual power plants as described in the Severe Accident Policy Statement, based in part on insights gained from risk analyses, can serve many useful purposes. For example, the search for risk outliers for individual plants should be performed. We believe that detailed qualitative information on plant characteristics and behavior is an important result of such a search, but that quantitative information (such as core melt frequency estimates for an individual plant) developed by a PRA is less robust. We are convinced also that direct participation by managerial, engineering, and operational personnel in a systematic examination of their plant can provide them with valuable insights and understanding of plant behavior in abnormal situations.

The Safety Goal Policy should be used by the NRC Staff chiefly as a standard for judging the adequacy and appropriateness of regulations and regulatory practices to assist them in reaching decisions about regulatory requirements. However, to make the Safety Goal Policy usable for these purposes, development in two areas is needed. The multiple goals and criteria should be related more logically in a hierarchical structure, and the "sampling" of existing plants should be expanded beyond the work which served as a basis for NUREG-1150, "Reactor Risk Reference Document," Draft for Comment, dated February 1987.

HIERARCHICAL STRUCTURE OF THE SAFETY GOAL AND IMPLEMENTATION PLAN

Several goals and guidelines are included in the Safety Goal Policy. We believe that it would be useful to present these in a hierarchical structure to facilitate implementation of the Policy in a range of circumstances. The highest level would serve as the Commission's statement of intent in regulating nuclear power and could then be used in decisions about broad policy matters and general comparisons with other industrial and technological activities. The lower levels could then be used, as development is completed, by the NRC Staff in making specific regulatory decisions. Each subordinate level of the hierarchy should be consistent with the level above, but should be a more practical surrogate, representing a simplification or quantification of the previous level. Each surrogate should not be so conservative that it creates a de facto new policy. It should also provide a basis by which to assure that the Safety Goal Policy objectives are being met.

A recommended hierarchical arrangement of the multiple goals in the Policy Statement is presented below.

- Level One: This would be the pair of qualitative goals as stated in the Commission Policy Statement of August 4, 1986.

- Level Two: This would be the pair of quantitative health objectives as stated in the same Policy Statement.

- Level Three: This would be the previously proposed general performance guideline that the likelihood of a large accidental release should be less than 10E-6 per reactor year.

  If this general performance guideline is to serve as a surrogate for the Safety Goal Policy objectives, as proposed in our letter of April 15, 1986, it should represent a level of safety consistent with the Level One and Two goals. A definition of a large release as one that will lead to whole body doses of 5 or 25 rem to an individual at a plant boundary, as has been given some public mention, does not satisfy this criterion. Such a definition is so much more restrictive than the Level One and Two goals that it, in effect, establishes an alternative policy rather than serving as a more easily applied surrogate. We believe that this is a distortion of the intent of the Policy and suggest that a consistent definition of a large release would be one that, if it occured, would result in significantly larger whole body doses.

- Level Four: This level of the hierarchy would consist of three performance objectives to be relied on in ensuring that the safety of operating plants is consistent with the Level One, Two, and Three criteria. These objectives should be explicit enough that they could be used by the NRC Staff in making decisions about specific regulations and regulatory practices. Such objectives are described below.

(1) The first performance objective would be an expressi[illegible] the effectiveness of plant accident <u>prevention</u> systems. We have previously recommended a goal of "less than 10E-4 per reactor-year" for the mean core melt frequency "for all but a few existing plants." By core melt, we mean loss of assured core cooling which can result in severe core damage. There is an unquantified, but probably substantial, difference between the probability of loss of assured core cooling and the probability of the "core on the floor stage". The latter is more surely threatening to the health and safety of the public, but is also less likely.

In relating this performance objective to the risk-based Safety Goals, one will have to confront the difficult technical issues associated with the progress of a severe accident. This will not be easy, but a core melt probability objective is less useful at this level of the hierarchy if its relation to the ultimate objective is unclear. Core melt, as defined here, is an identifiable waypoint in the development of a severe accident.

(2) The second performance objective would be an expression of the effectiveness of the design of plant accident <u>mitigation</u> systems. Between core melt, as defined above, and challenge to containment, as normally understood, there are several stages at which the accident sequence may be arrested. A containment performance objective cannot be stated simply in terms of the Level Three probability of a large release and the probability of a core melt as discussed above.

We recommend that as a minimum the containment performance objective should be such that there is less than one chance in ten for a large release for the entire family of core melt scenarios.

(3) The third performance objective would be an expression of how well the plant is operated. This remains to be developed. A separate objective of this sort would not be necessary if operating performance were appropriately considered in the first two performance objectives. However, present methods of analysis for performance objectives are based primarily on system design only. For this reason it seems necessary at this time to consider operations in a separate objective, if the Safety Goal

Policy is to be applied to plant operation and not just to plant design. We recognize this to be a major undertaking, but regard it as essential to a meaningful implementation of the Safety Goal Policy.

° Level Five: The final level of the Safety Goal Policy logic is the existing body of regulations and regulatory practices. Implementation of the Safety Goal Policy, as we propose, can be viewed as a review of "Level Five" to ensure that it is consistent with and carries out the intent of the goal levels above it. The overall policy implementation that we propose consists, in effect, of transforming a bottom-up system of regulation to a top-down system as the maturing of the nuclear industry and regulation and understanding of risk have permitted. In the end, as the effectiveness of the deterministic regulations and regulatory practices is more closely related to the Safety Goal Policy, it will be appropriate to adjust the regulations and regulatory practices to make them consistent with the Safety Goals.

SAMPLING OF PLANTS

As indicated above, our recommendation for implementation of the Safety Goal Policy is that it should be used principally as a measure of the adequacy of the regulations and regulatory practices. Safety performance at nuclear power plants then should reflect to a substantial degree the success of these procedures. Further in order to measure effectiveness of the regulations and regulatory practices, the product must be tested. The essential difference between what the ACRS proposes and what the Staff has proposed in this regard is that we believe the measurement of specific plants against the safety goal should be explicitly recognized as a sampling process. The goal of the process should be to determine why and how the regulations and regulatory practices have caused an individual plant or a class of plants to conform with or fall short of the goal, not to simply determine whether an individual plant or class of plants conforms with or falls short of the goal. The purpose of the body of regulations and regulatory practices should be to provide a population of nuclear power plants that conforms to the Commission's safety policy intent, as expressed by the Safety Goal Policy.

A Safety Goal Policy implementation plan structured as suggested above can and should be used by the NRC Staff in its evaluation of proposed changes in regulation that arise from a variety of sources, such as operating experiences and resolution of generic issues. However, we believe a more proactive program should be undertaken as part of the Policy implementation. This would be a prioritized, systematic review of the body of regulations and regulatory practices (i.e., Level Five of the recommended Safety Goal Policy hierarchy) for conformance with the overall Policy. Such a program would be, in some regards, a continuation of the work that has resulted in the draft NUREG-1150 and in previous assessments of full-scope PRAs for particular plants. However, we believe a new program can be better focussed on sampling a sufficient number of plants and classes of plants with the aim of assessing the effectiveness of regulations and regulatory practices that have guided the design, construction, location, and operation of these plants.

LIMITATIONS

We note that there must be recognition of important limitations in the implementation of the Safety Goal Policy. These limitations are essentially those of the PRA methodology used, and are caused by a fundamental inability to accurately predict and calculate precise values of risk. Variability in data, uncertainty about applicability of data, imperfect understanding of important physical phenomena, and inevitable incompleteness in analysis all contribute to this limitation.

The NRC Staff must recognize the limitations of risk analysis and limitations in the definition of the Safety Goals themselves and must apply sufficient margins within its regulations and regulatory practices to accommodate these limitations. They have always had to make such judgments and allowances. The key point is that the NRC Staff and the industry will be better able to make balanced and consistent decisions about regulation, design, and plant operation with guidance provided by the Safety Goals and PRA than without.

The development of a Safety Goal Policy has been a long and difficult, but an important and pioneering, effort. We believe an implementation plan along the lines we have proposed will ensure that the Policy is used effectively in regulation.

Additional comments by ACRS Member David Okrent are presented below.

Sincerely,

WKerr

William Kerr
Chairman

Additional Comments by ACRS Member David Okrent

1. The general plan proposed by the ACRS for implementation of the Safety Goal Policy seems attractive at first sight. The ACRS recommends the use of safety goal criteria to judge the adequacy of regulations rather than to make regulatory judgments about specific plants. The ACRS does not believe that PRA is sufficiently developed to be used to make narrowly differentiated decisions about specific plants.

   One major problem with this approach, in my opinion, arises from the current USNRC backfitting rule, which says in 50.109 Part (3):

   > The Commission shall require the backfitting of a facility only when it determines, based on the analysis described in paragraph (c) of this section, that there is a substantial increase in the overall protection of the public health and safety or the common defense and security to be derived from the backfit and that the direct and indirect costs of implementation for that facility are justified in view of this increased protection.

   A host of more easily specified backfits was hurriedly required in the United States after the accident at Three Mile Island, some with inadequate evaluation. However, the more difficult but often more significant issues have been deferred for years. These have now met up with a backlash against backfitting where I fear the pendulum has swung too far.

   France and the Federal Republic of Germany (FRG) have each maintained a disciplined program of backfitting, as well as promulgating safety improvements for nuclear plants to be built. France and the FRG have utilized PRA methodology in a way which resembles the ACRS proposal. Deterministic requirements were developed, frequently with the aid of insights obtained from PRA, to deal with perceived vulnerabilities in the overall safety approach. Cost/benefit analysis was not ignored but does not appear to have had a dominant impact on the decision-making process in France and the FRG.

   I would have more hope for the proposed ACRS approach if the basic USNRC safety position was to achieve in timely fashion a reasonable assurance that the high level safety goals and quantitative design objectives were being approached or met, without undue emphasis on cost/benefit analysis and the test for "a substantial increase in the overall protection" as has been practiced during the past few years.

2. I agree with the ACRS that an additional sampling of nuclear plants is needed in order to consider the adequacy of current regulations. In my opinion, draft NUREG-1150 not only is inadequate for this purpose, it is also misleading with regard to the present level of safety of LWRs in the United States, and should not be used by the NRC to provide such a perspective. I have many reasons for this opinion, some of which follow:

   ° External events are not included in draft NUREG-1150 (Reference 3).

   ° Many other potentially important contributors to risk, such as design and construction errors, aging and inadequate qualification of equipment, certain aspects of human error, certain types of systems interactions, and the effect of poor management quality are also not included in draft NUREG-1150.

   ° Some of the plants studied in draft NUREG-1150 had previously received the benefit of safety improvements resulting from one or two earlier PRAs on the same plant. This is not the case for the majority of operational LWRs.

   ° The PRAs in draft NUREG-1150 do not account adequately for the kinds of significant events which have occurred during the past two years or so at Surry, Brunswick, Trojan, Davis Besse, Rancho Seco, and TVA, among others.

   ° The PRAs in draft NUREG-1150 report core melt frequencies much smaller than those estimated for many of the plants examined in connection with USI A-45.

   Hence, not only is much additional sampling needed, but also some means must be developed for factoring into policy decisions the uncertainties and the significant gaps which exist in current PRAs, and for providing confidence that nuclear plants are operating within an envelope established by the Safety Goals and the supplementary objectives.

3. In view of the uncertainties and imprecision in PRA results, I disagree with the ACRS position that, if the general performance guideline on large releases is to serve as a surrogate, it should represent a level of safety consistent with the

Safety Goals. First I should note I do not look upon the general performance guideline on the frequency of a large release as wholly or primarily a surrogate. Furthermore, I prefer that one seek some level of assurance via performance guidelines that successively higher level goals or objectives will be met.

I question the suitability of the definition of a large release which is currently proposed by the NRC Staff. If a complementary cumulative distribution function of one or more early fatalities has a chance of 10E-6 per reactor year, may not the chance of 100 early fatalities be uncomfortably high at sites with large nearby population densities? Also, does such a proposed definition of a large release allow adequately for severe radioactive contamination of large land areas and for other relevant factors?

References:
1. Memorandum dated January 2, 1987 from Victor Stello, Jr., Executive Director for Operations, to the Commission, Subject: "Safety Goal Implementation Status," with enclosures on Framework for Safety Goal Implementation, Implementation of Safety Goals in Decisionmaking for Changing Generic Requirements, and Central Issues Treated in the Safety Goal Implementation Framework
2. U.S. Nuclear Regulatory Commission Report, 10 CFR Part 50, "Safety Goals for the Operations of Nuclear Power Plants," Policy Statement, dated August 4, 1986
3. U.S. Nuclear Regulatory Commission Report, "Reactor Risk Reference Document," NUREG-1150, Volumes 1 to 3, Draft for Comment, dated February 1987.

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

July 15, 1987

The Honorable Lando W. Zech, Jr.
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Zech:

SUBJECT: ACRS COMMENTS ON DRAFT NUREG-1150, "REACTOR RISK REFERENCE DOCUMENT"

During the 327th meeting of the ACRS, July 9-11, 1987, we discussed the draft report NUREG-1150, "Reactor Risk Reference Document," which was issued for comment in February 1987. The ACRS Subcommittee on Severe Accidents considered this report during meetings on January 29 and May 1, 1986 and the ACRS Subcommittees on Severe Accidents and Probabilistic Risk Assessment continued the review on June 3 and July 8, 1987. In our review we had the benefit of discussions with the NRC Staff and its consultants from Sandia National Laboratory (SNL). We also had the benefit of the documents referenced.

NUREG-1150 describes probabilistic risk analyses (PRAs) of several operating nuclear power plants. Results of PRAs for two of these were previously reported in WASH-1400. The plants analyzed had different containment types and included both PWRs and BWRs. The analyses are said to be "risk re-baselining"; i.e., the methods used are current, the data used in the analyses include both generic and plant-specific information, the computations make use of codes that have been developed since the publication of WASH-1400, and the risk calculations make use of the so-called Source Term Code Package (STCP) that includes much of the information developed by the NRC research program on severe accidents (although what was used was a slightly modified version of the published STCP). Containment performance is treated in much more detail in NUREG-1150 than it was in WASH-1400.

In addition to calculations of risk attributed to internal initiators, this report describes the results of studies which attempted to predict the uncertainties in the predictions of a number of relevant quantities, including core melt frequency and the probabilities of early and delayed fatalities.

In assessing public risk, the current version of NUREG-1150 is incomplete, since external accident initiators are not treated and, based on results from other PRAs, these may produce significant contributions to risk. Work has begun on external initiators, and later versions of the report will contain the results.

The report and its supporting documents are voluminous, and the amount of information reported is almost overwhelming. However, we believe the significance of the results and the anticipated use of the information in the regulatory process should be made explicit.

Among the conclusions reported, the following appear to be significant:

(1) The report concludes that, for the plants examined, the risk contributors are sufficiently disparate that no general conclusions can be drawn concerning the risk of plants not examined.

(2) The calculated risk of each plant analyzed was less than the quantitative health effects objectives in the Safety Goal Policy Statement. However, as mentioned above, the calculated risk did not include contributions from external initiators.

(3) The calculated risks for Surry, Unit 1, and Peach Bottom, Unit 2, were not markedly different from those reported in WASH-1400. We were told that a number of risk-reducing improvements had been made for these plants since the original analysis, but that these were somewhat offset by newly discovered risk contributors.

One of the original aims of the work reported in NUREG-1150 was to determine if an analysis of these selected plants would permit conclusions to be drawn concerning the risks of other operating plants not analyzed. So far as we can determine from the report and from discussions with the Staff, their conclusion is that these plants (and other plants that have been the subject of PRAs) are sufficiently different, and the risk contributors are sufficiently diverse, that little can be learned about one plant from the analysis of another plant, even when they are of the same general type.

This conclusion is both surprising and disturbing. If correct, it raises serious doubts about the breadth of application of these efforts. The Staff has not provided convincing reasons for this conclusion. More effort is needed to determine why this conclusion should be accepted, because such a conclusion would have far-reaching consequences for several Commission policies.

We have the following additional comments:

(1) We are skeptical of the method by which expert opinion was used in predicting uncertainties. Explanation of and justification for the

method are obscure. There is also reason to believe that the way in which the method is used can have a significant influence on the uncertainties predicted. It is thus almost impossible to interpret the significance of the reported uncertainties or to subject them to peer review.

(2) Many of the codes used in the calculation are relatively new. The validity of several of the codes is not well established. Furthermore, many of them have not been published and are not yet available to people outside the national laboratories. Serious peer review of the results reported is thus almost impossible.

(3) It was emphasized by the Staff that a major contribution of the report was the "insights" provided. We recommend that these insights be better identified and that their significance for those who are not PRA practitioners be made more clear.

(4) Human performance contributions to risk (both positive and negative) are not well described by PRAs. This report does not correct that deficiency.

(5) We were told by the Staff that, in light of insights developed during the work reported, resolutions or proposed resolutions of a number of Unresolved Safety Issues are to be revisited. We recommend that, as an aid to understanding the report, these instances be identified. We recommend also that the interaction between those responsible for the resolution of Safety Issues and those responsible for this report be improved.

One might conclude, both from the report and from comments made by the Staff, that the NRC regulations are inadequate to determine plant equipment and procedures necessary to protect public health and safety. If this is the Staff's conclusion, it is a dramatic finding and should be emphasized, and the position developed more effectively than it is in the present draft. If, however, regulations can be used as a mechanism to protect public health and safety, and we believe they can, we recommend that the Office of Nuclear Reactor Regulation begin early examination of this report, both to apply its insights and to guide its further development.

Sincerely,

W Kerr

William Kerr
Chairman

References:
1. U.S. Nuclear Regulatory Commission, NUREG-1150, Reactor Risk Reference Document, Volumes 1, 2 and 3, Draft issued for comment, dated February 1987
2. U.S. Nuclear Regulatory Commission, NUREG-75/104, "Reactor Safety Study - An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," dated October 1975 (formerly issued as AEC report WASH-1400).
3. U.S. Nuclear Regulatory Commission, NUREG/CR-4587, "Source Term Code Package: A User's Guide (MOD1)," Battelle Columbus Laboratory, dated July 1986.

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

June 9, 1987

Mr. Victor Stello, Jr.
Executive Director for Operations
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Stello:

SUBJECT: ACRS COMMENTS ON PROPOSED REVISED STANDARD REVIEW PLAN SECTION 3.6.2, "DETERMINATION OF RUPTURE LOCATIONS AND DYNAMIC EFFECTS ASSOCIATED WITH THE POSTULATED RUPTURE OF PIPING," DATED OCTOBER 2, 1986

In our letter to you dated November 12, 1986 concerning NRC Staff-proposed revisions to Standard Review Plan (SRP) Section 3.6.2, "Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping," we recommend that existing SRP requirements for postulating break sizes and locations should be retained where they relate to establishing compartment and subcompartment pressure buildup, particularly outside the primary containment.

We continue to put forth this recommendation but see no problem with the issuance of the revised SRP Section 3.6.2, provided the guidance stated in Mechanical Engineering Branch Technical Position 3-1, subpart B.1.c.(4) is implemented. This guidance states:

> If a structure separates a high energy line from an essential component, that separating structure should be designed to withstand the consequences of the pipe break in the high-energy line which produces the greatest effect at the structure irrespective of the fact that the above criteria might not require such a break location to be postulated. [The "above criteria" are the criteria for postulating high-energy fluid systems pipe rupture in areas other than containment penetrations.]

The retention of this general provision should assure adequate protection of essential components against pipe whip, jet impingement, and the pressurization effects of high-energy line ruptures inside of compartments and subcompartments, even after the arbitrary intermediate breaks are eliminated. We believe this adequately complies with our recommendation and therefore approve publication of the Federal Register Notice on the revised SRP 3.6.2.

Sincerely,

WKerr

William Kerr
Chairman

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

June 9, 1987

Mr. Victor Stello, Jr.
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Stello:

SUBJECT: ACRS COMMENTS ON PROPOSED REVISIONS TO STANDARD REVIEW PLAN SECTIONS 6.5.2, "CONTAINMENT SPRAY AS A FISSION PRODUCT CLEANUP SYSTEM" AND 6.5.5, "SUPPRESSION POOLS AS FISSION PRODUCT CLEANUP SYSTEMS"

During its 326th meeting, June 4-6, 1987, the members of the ACRS discussed the proposed changes to Sections 6.5.2, "Containment Spray as a Fission Product Cleanup System," Revision 2 and 6.5.5, "Suppression Pools as Fission Product Cleanup Systems," Revision 0, of the NRC Standard Review Plan. These matters were also discussed during a meeting of our Nuclear Plant Chemistry Subcommittee on May 19, 1987.

As a result of these discussions, we endorse the general approach being proposed by the NRC Staff.

Sincerely,

WKerr

William Kerr
Chairman

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

August 12, 1987

Mr. Victor Stello, Jr.
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Stello:

SUBJECT: ACRS ACTION ON THE PROPOSED SECTION 3.6.3, "LEAK-BEFORE-BREAK EVALUATION PROCEDURES," OF THE NRR STANDARD REVIEW PLAN

During the 328th meeting of the ACRS, August 6-8, 1987, we discussed the proposed Standard Review Plan (SRP) Section 3.6.3, "Leak-Before-Break Evaluation Procedures," which is intended to provide detailed guidance on the implementation of the revised provisions of General Design Criterion 4, Environmental and Missile Design Bases (GDC-4) of Appendix A to 10 CFR Part 50.

As a result of this discussion, we concur in the NRC Staff's proposal to issue the proposed SRP section for public comment. After the public comment period, we expect to review the proposed final version of this SRP section together with the public comments and the NRC Staff's response to them.

Sincerely,

WKerr

William Kerr
Chairman

Reference:
Memorandum to Distribution from R. J. Bosnak, DD/DE/RES, Subj: Resolution of ACRS/CRGR Comments on Final Broad Scope Amendment to GDC-4, dated July 29, 1987, with Enclosures:

1. Memorandum for Commissioners from Victor Stello, Jr. EDO, Subj: Final Broad Scope Rule to Modify General Design Criterion 4 of Appendix A, 10 CFR Part 50

2. Standard Review Plan, NUREG-0800 (Formerly NUREG-75/087) 3.6.3, Leak-Before-Break Evaluation Procedures

cc: S. Chilk, SECY
R. Bosnak, RES
R. Hernan, NRR

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

August 12, 1987

Honorable Lando W. Zech, Jr.
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Zech:

SUBJECT: ACRS COMMENTS ON PROPOSED FINAL BROAD SCOPE RULE TO MODIFY GENERAL DESIGN CRITERION 4, ENVIRONMENTAL AND MISSILE DESIGN BASES (GDC-4)

During the 328th meeting of the ACRS, August 6-8, 1987, we met with representatives of the NRC Staff and reviewed the final broad scope rule to modify GDC-4. The ACRS Subcommittee on Metal Components held meetings on this subject on February 27-28, 1986 and July 24, 1987 with representatives of the NRC Staff and the nuclear industry. We also had the benefit of the document referenced.

We endorse the issuance of the proposed final rule modifying GDC-4 regarding consideration of the dynamic effects of postulated pipe ruptures in a nuclear power plant's design basis. The acceptance criteria outlined by the NRC Staff appear to be conservative enough to ensure that the pipes in question will leak at easily detectable rates well before complete breaks occur.

The proposed rule states, "the Commission will permit applicants and licensees to justify alternative environmental qualification requirements case-by-case to replace those environmental qualification requirements which were associated with postulated pipe ruptures ...." We wish to be kept informed of any relaxation of environmental qualification requirements outside primary containment which are based on leak-before-break consideration.

Further, we have an interest in the possibility that a licensee may be able to demonstrate that water hammer is unlikely to occur in a given high energy system outside of primary containment, and therefore leak-before-break concepts can be applied. Should such a situation be proposed by an applicant or licensee and found acceptable to the NRC Staff, we wish to be kept informed.

Additional comments by ACRS Members David Okrent and Glenn A. Reed are presented below.

Sincerely,

WKerr

William Kerr
Chairman

Additional Comments by ACRS Members David Okrent and Glenn A. Reed

Despite the seeming arrival of cures from time to time, we believe that the long history of stress corrosion cracking in BWRs, and the absence of an adequate history of operating experience free from intergranular stress corrosion cracking (IGSCC) make it prudent not to permit application of "leak-before-break" to BWR high energy piping at this time, even if stress improvement and improved water chemistry are present.

Reference:
Memorandum to Distribution from R. J. Bosnak, DD/DE/RES, Subj: Resolution of ACRS/CRGR Comments on Final Broad Scope Amendment to GDC-4, dated July 29, 1987, with Enclosures:

1. Memorandum for Commissioners from Victor Stello, Jr. EDO, Subj: Final Broad Scope Rule to Modify General Design Criterion 4 of Appendix A, 10 CFR Part 50

2. NUREG-0800 (Formerly NUREG-75/087), Standard Review Plan, Section 3.6.3, "Leak-Before-Break Evaluation Procedures"

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

December 8, 1987

Mr. Victor Stello, Jr.
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Stello:

SUBJECT: ACRS COMMENTS ON THE REVISED ASB 3-1 OF STANDARD REVIEW PLAN 3.6.1, REQUIREMENTS FOR ARBITRARILY POSTULATED JET IMPINGEMENT EFFECTS IN THE BREAK EXCLUSION ZONE

In the referenced memorandum, Eric S. Beckjord, Director, Office of Nuclear Regulatory Research, has requested approval of a Federal Register notice that would eliminate from the design basis the jet impingement effects associated with the arbitrary one-square-foot break in the break exclusion (superpipe) zone of main steam and feedwater lines outside the containment.

Based on the draft Federal Register notice and discussions with the staff, we understand that the arbitrary one-square-foot break will be retained for environmental qualification of essential equipment and for structural pressurizations. Moreover, we understand that the staff will continue to enforce separation and isolation of essential equipment as the preferred method of providing protection, without referring to postulated jet impingement effects.

Under these conditions, we endorse publication of the draft Federal Register notice revising ASB 3-1 of Standard Review Plan 3.6.1.

Sincerely,

W Kerr

William Kerr
Chairman

Reference:
Memorandum dated October 2, 1987 for Edward L. Jordan, Chairman, CRGR and William Kerr, Chairman, ACRS, from Eric S. Beckjord, Director, RES, Subject: Request for Approval to Publish a Federal Register Notice Revising ASB 3-1 of SRP 3.6.1.

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

February 11, 1987

The Honorable George H. W. Bush
President of the 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 submits herewith its comments on the Nuclear Regulatory Commission's Safety Research Program for Fiscal Year 1988.

We note that the trend of continually decreasing funding levels for the NRC Safety Research Program over the past several years has been arrested and even slightly reversed. We are heartened by this development and hope that it persists.

Sincerely,

W Kerr

William Kerr
Chairman

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

February 11, 1987

The Honorable James C. Wright, Jr.
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 submits herewith its comments on the Nuclear Regulatory Commission's Safety Research Program for Fiscal Year 1988.

We note that the trend of continually decreasing funding levels for the NRC Safety Research Program over the past several years has been arrested and even slightly reversed. We are heartened by this development and hope that it persists.

Sincerely,

W Kerr

William Kerr
Chairman

REVIEW AND EVALUATION OF THE
NUCLEAR REGULATORY COMMISSION
SAFETY RESEARCH PROGRAM
FOR FISCAL YEAR 1988

A REPORT TO
THE CONGRESS OF THE UNITED STATES OF AMERICA

BY

THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
U.S. NUCLEAR REGULATORY COMMISSION

WASHINGTON, D.C. 20555
FEBRUARY 1987

INTRODUCTION

Our comments below relate to the proposed program-support funding of $103.6 million for the FY 1988 NRC Safety Research Program. These comments are limited chiefly to research that is not being proposed but should be, or in some cases, to research that is proposed but may not be needed.

1. REACTOR SYSTEM SAFETY

1.1 Thermal-Hydraulic Transients

1.1.1 General Comments

Research planned in the thermal-hydraulic area is divided into two general parts. The first is a comprehensive program to improve the understanding of thermal-hydraulic behavior in Babcock and Wilcox (B&W) reactor systems; in its fullest form, this calls for substantial industry support. The second is a more general program of code development and experimental work that does not include direct industry participation. We believe that the emphasis on B&W systems is appropriate and that the industry should provide major support for this program, as has been proposed by the Office of Nuclear Regulatory Research (RES) Staff. It is also important that the NRC maintain a viable program of thermal-hydraulic research into the foreseeable future, including integral testing, separate-effects testing, and code development.

Our comments on the proposed research in the thermal-hydraulic area are given below.

1.1.2 Integral Facilities

RES has developed a comprehensive research plan to address the technical issues and regulatory needs associated with the thermal-hydraulic performance of B&W plants. Central to this plan is the proposal to construct new integral test facilities. Optimum and timely results from these facilities will depend upon financial support from the industry. We support this plan.

1.1.3 Separate Effects

Our February 19, 1986 report to the Congress included the recommendation that the NRC study the complicating effects that water hammer may have on thermal-hydraulic transients. The NRC and the industry have now initiated a cooperative effort in this regard. We support this effort and expect to monitor its progress. Funding levels appear adequate at this time.

The NRC has initiated a program to develop an information base for the complex thermal-hydraulic phenomena involved with the "bleed and feed" process used either to remove core decay heat or to allow controlled depressurization of the primary coolant of a pressurized water reactor (PWR) plant. We believe that subsequent to the development of the information base RES should allocate funding for any additional research found necessary in this area.

1.1.4 Code Development

The NRC has developed a revised Emergency Core Cooling System (ECCS) Rule (10 CFR Part 50, Appendix K - ECCS Evaluation Models) that allows the use of realistic (best estimate) evaluation models to calculate the effects of loss-of-coolant accidents (LOCAs). However, the acceptability of realistic models rests on the development of satisfactory methodology to determine the overall uncertainty associated with these models. A related effort which addresses code applicability and scaling studies also is necessary; such work is either ongoing or planned by RES. We strongly support these efforts. In particular, funding should be assured over the next few years sufficient to allow NRC to obtain the necessary test and analytical data, primarily through the International Code Assessment Program and cooperative international efforts such as the 2D/3D Program.

1.2 Accident Evaluation

The Accident Evaluation Research Program being proposed includes a significant experimental component. We believe that the relationship of this research to the severe accident regulatory issue should be made clearer than it now is. For example, there are three major experimental programs to investigate phenomena that will be encountered (if at all) only after the reactor core has melted and has penetrated the vessel. The experiments are related to containment heating, to core-concrete interaction, and to containment behavior under extreme overpressure.

These areas of research all bear directly on issues relating to containment capability and containment failure modes. Each of these research areas is said to be designed to reduce some of the uncertainties identified in NUREG-0956, "Reassessment of the Technical Bases for Estimating Source Terms." The magnitude of these uncertainties is being estimated in the course of the preparation of NUREG-1150, "Reactor Risk Reference Document." However, in order to plan properly additional research to reduce uncertainties, it would appear that the Office of Nuclear Reactor Regulation (NRR), or some group, should first consider what uncertainties can be tolerated in connection with its regulatory responsibilities. We have not seen evidence of this consideration. We, and we suppose RES as well, must, under the circumstances, try to judge the relevance of the proposed research with insufficient information.

With this caveat we make the comments noted below. These comments do not represent a complete coverage of the Accident Evaluation Research Program. We use them as an example of our conclusion that more consideration needs to be given to what has been learned from the research of the past five years or so, and what uncertainty can be tolerated by the regulators; or, put another way, what are the questions that regulators have encountered or are likely to encounter in dealing with severe accidents that cannot be answered with existing information. With diminishing resources, it is increasingly important that the research be specifically designed to address safety concerns.

* The research on containments under extreme overpressure seems well designed and should produce results that are relevant, and that will contribute to the calibration of codes being developed for a description of containment behavior.

* The work on core-concrete interaction is probably needed, but a more detailed examination of the ways in which this may affect containment failure, as well as the uncertainties attributable to incomplete understanding of the complex phenomena that characterize this interaction, would make it more likely that the research to be done would answer questions that will be encountered by NRR.

* Some risk analyses conclude that high-pressure-core-melt sequences may produce enough direct heating of the containment atmosphere to cause early containment failure by overpressure. The proposed research on direct containment heating may be of considerable interest in understanding the interaction of small particles of molten metal with containment atmospheres. However, it is relevant to reactor safety only if sequences which could produce such small particles in PWRs have a sufficiently high probability. It is our opinion that equal effort should be devoted to establishing the likelihood and the effect of direct heating events, since absent such work, the program may well be misdirected. We also recommend that the experimental investigation give first consideration to the possibility of atomizing the required amount of material as well as the effects of containment geometry, and to the presence of water in the subvessel cavity.

1.3 Risk and Reliability

The funding for the Risk and Reliability Research Program continues to be directed away from the development of risk assessment methodology and toward applications. The work in the applications area for the most part consists of software development and plant-specific risk assessment. However, we see some danger that with the current budget constraints, work on applications in support of licensing efforts will be undertaken at the expense of developmental work. It is important that the developmental work continue.

We recommend that the applications work be coordinated with the eventual user by a systematic process that would encourage the user to be involved in the development of the work product. RES has taken steps to involve the industry and individual licensees in this work. We consider it appropriate that some of this applications effort should be done by the industry or individual licensees rather than by the NRC.

In our previous reports to the Congress and to the Commission, we recommended that:

- The completeness of the current family of plant-specific probabilistic risk assessments (PRAs) should be examined by continued search for possible weaknesses in current probabilistic analyses, e.g., accident paths either not currently evaluated or dismissed as insignificant, which may, on closer scrutiny, prove to be very important to risk.

- An improved evaluation of the entire family of containment designs should be performed.

- Improved methods for factoring uncertainty into decision making should be developed.

The NRC Staff is currently developing methods for implementing the Severe Accident Policy and the Safety Goal Policy. We believe that difficulties are being encountered in part because of the lack of important answers in areas such as those listed above, as well as in the treatment of external events, environmental effects, aging, management quality, and human errors. We recommend that RES initiate additional work in such areas by reprogramming funds in the FY 1987 budget and continue to support this work in FY 1988. If necessary, this should be accomplished by deferring the applications work or finding other sources of support for this work.

1.4 Human Factors

In its recent report to the NRC, "Revitalizing Nuclear Safety Research," the National Research Council's Committee on Nuclear Safety Research has joined the ACRS in criticizing the NRC for not performing any research in the human factors area. A specific study to determine the need for and nature of human factors research is being performed by the National Research Council's Committee on Human Factors. The final report of this Committee is to be provided to the NRC during July 1987. We anticipate that this report will include recommendation for a considerable program of research effort in the human factors area. Plans for beginning such a program should be factored into the budget by RES.

2. ENGINEERING SAFETY

2.1 Electrical Equipment Qualification

No funding for electrical equipment qualification research has been proposed for FY 1988. In our February 19, 1986 report to the Congress, we recommended that this research be continued; contrary to this recommendation, the research was terminated by the NRC at the end of FY 1986.

The objective of this research is to assess the probability of survival and performance of aged electrical equipment when subjected to hostile environmental conditions during and following incidents, including severe accidents, fires, hydrogen burns, seismic events, and credible combinations. The results obtained from this research are important to prevent accidents as well as to mitigate the consequences of accidents, should they occur. We consider the continuation of this work to be vital to the nuclear safety program and again recommend that it be reinstituted and adequately funded.

Four unique test facilities, with a combined cost of over $2 million, were constructed at Sandia especially for this program. To preserve and maintain the existing test facilities and staff experience sufficient to continue and complete the electrical equipment qualification work efficiently, it will be necessary to continue funding this work in FY 1987. Since its inception (about 1976), more than $10 million has been spent on this program. Funds needed to continue and complete the program are approximately $1.5 million in FY 1987 and $0.9 million in FY 1988.

2.2 Effects of Earthquakes on Operating Plants

In our previous reports, we urged that RES establish an integrated program in this area and coordinate closely this work with NRR and ongoing industry work. We believe that this has been accomplished effectively in the Seismic Safety Research Program. We believe that the program is well managed and will, in the near future, produce answers that will help to resolve important issues.

2.3 Primary System Integrity

Cast stainless steel components in the primary system lose ductility with time in service. The implications of this phenomenon for long-term primary system integrity are significant. Appropriate emphasis should be placed on ascertaining the likelihood of flaws resulting from fabrication or from service, and on developing means of assessing conditions under which they could pose significant risk.

3. WASTE MANAGEMENT

3.1 High-Level Waste

Although we are satisfied with most aspects of the research program on high-level wastes, we believe that more attention needs to be directed to related work under way in other countries. This extends beyond the formal agreements for cooperative research that the NRC Staff has developed with selected foreign groups. The Staff should also keep abreast of activities of the Nuclear Energy Agency, Organization for Economic Cooperation and Development, relative to the development of guides for demonstrating, both directly and indirectly, the capabilities of a repository for assuring the safe retention of radioactive wastes.

3.2 Low-Level Waste

We believe that more effort should be directed to studies that will assist the states in ranking and selecting the most appropriate disposal systems for low-level wastes, based on the nature and characteristics of the sites available and associated technological and economic considerations.

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

April 13, 1987

The Honorable Lando W. Zech, Jr.
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Zech:

SUBJECT: ACRS REPORTS TO THE NRC ON THE RESEARCH PROGRAM

In our February 19, 1986 report to the Congress on the NRC safety research program, we requested reconsideration of the statutory requirement for an annual report. We now wish to propose for your concurrence a change in the nature and timing of the advice we provide to the Commission on the research program.

As required by Section 29 of the Atomic Energy Act of 1954, as amended by Section 5 of Public Law 95-209, we have submitted reports to the Congress since 1977. The first was submitted in December 1977, but subsequent reports have been submitted in February of each year after the budget has been received by the Congress. In 1979, the Commission requested a similar report, to be submitted each year at the time the RES budget was being prepared and considered by the Commission.

The scope and content of these reports have changed markedly over the years. Prior to 1986, our report to the Congress was a NUREG document of 40 to 60 pages containing relatively detailed comments on many aspects of the research program, comments that were directed more to the NRC Staff than to the Congress. Our last two reports to the Congress have been brief 5 or 6 page letter reports, with correspondingly less detail and based to a considerable extent on our earlier report to the Commission on the same budget. Our reports to the Commission also were reduced in scope, beginning in 1982, following an exchange of correspondence between then Chairman Palladino and ACRS chairmen (Attachments 1-4).

Some recent actions or developments that suggest a need to reexamine our role in relation to the NRC research program are discussed below.

Your letter of September 18, 1986 to D. A. Ward suggested that it would be useful for the committee to "Advise the Commission on the effectiveness and correctness of direction of NRC's research program to ensure that research is relevant to the agency's safety mission." We do not believe that our present annual reports are responsive to this need, even in their current abridged form. Nor is it clear that you expect such advice on an annual basis, and if so, on the current schedule based on the budget process.

In its December 8, 1986 report to the Commission entitled, "Revitalizing Nuclear Safety Research," the National Research Council Committee on Safety Research recommended that the NRC should empanel an independent advisory group "...charged with independently reviewing for the director of research, from the perspective of the general principles cited in this report, the overall structure and thrust of the research program." In SECY-87-52, "Independent Advisory Panel for the Office of Research," the EDO has recommended the creation of a Standing Board of the National Research Council to perform that review.

There has been a significant reduction in resources available to the ACRS, and a further reduction is proposed. Even in the abridged form of the last few years, each of the two annual reports (one for the Congress and one for the Commission) requires two meetings of the rather large Safety Research Program Subcommittee, several hours of full committee time, and substantial amounts of review by ACRS technical or generic subcommittees to obtain information and develop positions on specific portions of the research program.

In view of the developments mentioned above, we believe that we can best serve the Commission by reporting to you on the effectiveness and correctness of direction of those elements of the safety research program that appear to either you or us to warrant attention at any given time. In addition, we would think it appropriate, from time to time, to provide some perspective, not tied to specific issues, on the overall thrust and relevance of the program. These reports would not be submitted on a schedule related to the budget process nor even on a strictly annual basis. They would be intended to keep you informed of our views on the program and to provide you with a basis for formulating and defending the research program and budget.

We will be happy to discuss this with you and the other Commissioners.

Sincerely,

WKerr

William Kerr
Chairman

Attachments:

1. Letter from J. Carson Mark, Chairman, Advisory Committee on Reactor Safeguards, to Nunzio J. Palladino, Chairman, U.S. Nuclear Regulatory Commission, Subject: ACRS Review and Reports on Safety Research Programs, dated October 20, 1981.

2. Letter from Nunzio J. Palladino, Chairman, U.S. Nuclear Regulatory Commission, to J. Carson Mark, Chairman, Advisory Committee on Reactor Safeguards, dated December 10, 1981.

3. Letter from J. Carson Mark, Chairman, Advisory Committee on Reactor Safeguards, to Nunzio J. Palladino, Chairman, U.S. Nuclear Regulatory Commission, Subject: ACRS Review and Reports on NRC Safety Research Programs, dated December 14, 1981.

4. Letter from P. Shewmon, Chairman, Advisory Committee on Reactor Safeguards, to Nunzio J. Palladino, Chairman, U.S. Nuclear Regulatory Commission, Subject: Procedures for ACRS Review of the NRC Long-Range Research Plan, dated June 7, 1982.

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

October 20, 1981

Honorable Nunzio J. Palladino
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: ACRS REVIEW AND REPORTS ON SAFETY RESEARCH PROGRAMS

Dear Dr. Palladino:

Since 1977, the ACRS has been required by the Congress to report to it annually on the NRC Safety Research Program. This report is prepared each year, after OMB has transmitted the budget request to the Congress in November, and is submitted in February before the appropriate Congressional committees complete their recommendations on the authorization bill.

Since 1979, we have provided a report to the Commission on the research program and its budget, usually just before the EDO budget goes to the Commission for final action in July. This report has been similar in scope to the Report to Congress, although the original request from the Commission was for comments on the budget rather than a complete review of the safety research program.

In 1981, we prepared a report to the Commission on the draft Long Range Research Plan (LRRP). This report was in the form of a letter rather than the format of the other two reports noted above. This report, too, was requested by the Commission, and existing procedures call for similar reviews and reports on the yearly updates of the LRRP.

We believe that our reviews of the safety research program in general, and of individual areas and projects, have been useful to both us and the RES Staff. We believe that the Staff has been responsive in large part to our comments and recommendations.

However, we do not believe that the benefits from our reviews and reports justify the expenditure of resources by the ACRS, its Staff and consultants, and by the RES Staff, that has been required to make three separate reviews each year and prepare three separate reports. We understand that Mr. Minogue agrees with this evaluation.

ATTACHMENT 1

We propose to ameliorate this situation, without reducing the extent or effectiveness of our review of the program and our interaction with the RES Staff, by the following procedures:

Report to Congress. We will continue to prepare this report, as before. It will be relatively long and relatively comprehensive, and will provide comments on the nature, scope and effectiveness of the program as well as on needs and proposed funding levels. This report will continue to be available in February, and thus can be used by the RES Staff as a basis for its update of the LRRP and its preparation of the next budget cycle.

Report to the Commission. If requested, we will, of course, provide comments or advice to the Commission on the RES budget request or on specific portions of the safety research program or on funding levels in detail or in general. However, we prefer not to provide evaluations and comments of the kind and scope already included in the Report to Congress. Such a report to the Commission would be brief and in letter form.

Long Range Research Plan. The first LRRP developed was little more than a five-year projection of current programs and current needs, and provided little to review in addition to the reviews we had already made of ongoing programs and those planned for the next one or two years. We believe, therefore, that reviewing the LRRP would not be an effective use of our time unless a more meaningful plan is developed.

We would be pleased to have your comments on these proposed changes in procedures, and we will be willing to discuss them with you and the Commissioners at your convenience.

Sincerely,

J. Carson Mark

J. Carson Mark
Chairman

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

CHAIRMAN

December 10, 1981

Dr. J. Carson Mark
Chairman
Advisory Committee on Reactor Safeguards
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Dr. Mark:

Your letter of October 20 on the ACRS's several annual reviews of the RES program highlights a concern over the amount of time and effort that both the ACRS and the RES staff expend on the reviews. RES has estimated that it expends on the order of 150 to 300 man weeks per year preparing for, participating in, and doing follow-up analysis for the ACRS reviews of its program and budget. Your letter clearly expresses concern for the amount of ACRS time also spent in these reviews. We share with you a desire to significantly reduce the time spent on these reviews, while at the same time not reducing the benefit that we and the Congress receive from the ACRS input and guidance to our research program.

We agree that ACRS should not have to perform three separate reviews each year. We concur with your proposed approach to develop a plan in which the Committee would conduct only one thorough review each year, with possibly the need for some updating by RES to keep you abreast of important changes. We agree with your recommendations regarding the report to the Commission on the RES budget request and the preparation of a comprehensive report to the Congress in February of each year. However, we believe that in view of the timing of the annual report to Congress on the Research program, it would also benefit the Commission, Congress and the RES staff if this review included consideration of the Long Range Research Plan (LRRP).

It is our intention that the annual development and refinement of the LRRP constitute the foundation for the planning of our research program. Preparation of the LRRP permits us to lay out directions of the research program for the coming years and to obtain user office endorsement of these program directions. At this time, the research programs are in the formative stage and are more amenable to guidance and advice than at any other stage. The planning effort can benefit from the perspective of a group of experts in nuclear safety problems who are in intimate contact with the current regulatory challenges. A thorough review by ACRS at this stage should provide all of the background and material needed to

ATTACHMENT 2

allow the fulfillment of your obligations to the Congress and would be sufficient to provide my fellow Commissioners and me the benefit of your advice for our review of the RES budget in the summer. Normally, the plan would be available for your review in December with sufficient time for you to hold subcommittee meetings in January. The report for the Congress could also include an update of any changes (mostly deletions) to the program for the budget year being considered by the Congress that may have occurred as a result of the NRC-internal and OMB budget review process.

The LRRP for 1983-1987 (NUREG-0740) was the first attempt at preparing a plan under the current criteria. (Five-year plans were prepared in 1976 and 1977.) RES has received constructive criticism on that plan; the next version, which is now being prepared, will have the benefit of that advice. We expect that it will include more detailed program descriptions, discussion of need and expected use of results.

Thus, we concur with the ACRS recommendations contained in your letter of October 20, 1981 with the exception that an ACRS review of the LRRP be included in the comprehensive review of the research program which forms the basis for your annual report to Congress. This would give us the benefit of your advice at the earliest and most productive stage and, we believe, result in the most efficient use of your and our RES staff time.

Commissioner Ahearne agrees to the ACRS reducing their level of budget review, but would have preferred to retain some level of ACRS review and comment on the more significant items. The report to Congress is on the budget submitted to Congress. This review does not duplicate the report to the Commission on the budget being considered for future submittal. However, given the nature of the Long Range Research Plan, he agrees to relieving ACRS of their review of the Long Range Research Plan.

Please let us know if you have any additional thoughts on this matter.

Sincerely,

Nunzio J. Palladino

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

December 14, 1981

Honorable Nunzio J. Palladino
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUBJECT: ACRS REVIEW AND REPORTS ON NRC SAFETY RESEARCH PROGRAMS

Dear Dr. Palladino:.

In our letter of October 20, 1981 we expressed our belief "that reviewing the LRRP would not be an effective use of our time unless a more meaningful plan is developed." Although we anticipate significant improvements in the LRRP, it is perhaps too late to use the new LRRP as a basis for our report to Congress on the FY 1983 program since that report is well under way, and we have not yet received the new plan. Nevertheless, we intend to review the plan and, to the extent needed and practicable, provide you and the Commissioners with our comments. It is likely that our comments this year can be based primarily on the reviews we have carried out in preparation for our report to Congress; extensive interaction with the RES Staff should not be necessary. Nevertheless, we will consider ways in which our review of the FY 1984 Safety Research Program can be carried out in order to provide you with timely and useful comments on the LRRP and, at the same time, provide us with the information and insights we need to prepare our report to the Congress.

With regard to a review and report to the Commission in July on the RES budget request, we said in our letter of October 20, 1981 that we will continue to provide comments on funding levels, in detail or in general, and on specific portions of the program. In doing so, however, we would expect to limit our interaction with the RES Staff; this would be possible if there is an easily identifiable relation between their budget request and the needs and programs described in the LRRP. Moreover, we would not intend to elaborate on the bases for our recommendations if it is possible to relate them to comments made previously in connection with the LRRP and our report to Congress.

We will continue to make both general and specific recommendations to the Commission and to the RES Staff. It would be helpful to us in our continuing review of the Safety Research Program, if RES would respond in writing to each recommendation, general or specific, made in our report to the Congress.

In summary, we believe that procedures can be developed to provide the information requested in your letter of December 10, 1981.

Sincerely,

J. Carson Mark

J. Carson Mark
Chairman

ATTACHMENT 3

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

June 7, 1982

The Honorable Nunzio J. Palladino
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Dr. Palladino:

Subject: PROCEDURES FOR ACRS REVIEW OF THE NRC LONG-RANGE RESEARCH PLAN

During the ACRS meeting with the Commissioners on June 4, 1982, you asked that we summarize our views on the preparation of an annual ACRS report on the Long-Range Research Plan (LRRP). The memorandum from S. Chilk, SECY, to W. J. Dircks, EDO, (COMJA 80-13) dated April 22, 1980 indicated that we should review and comment on an updated LRRP in February of each year. Accordingly, the first ACRS report, issued April 14, 1981, was on the draft LRRP for FY 1983-FY 1987 (NUREG-0740). Your letter of December 10, 1981 confirmed your interest in receiving ACRS comments on the LRRP, and we issued our second report on April 5, 1982, on the draft LRRP for FY 1984-FY 1988 (NUREG-0784).

We have found the LRRP to be useful; however, the review of the LRRP has involved considerable effort for both us and the NRC Staff. This effort is in addition to that involved in our annual reports on the NRC safety research program and budget to the Commission and the Congress.

We propose that we discontinue our formal report to the Commission on the LRRP. However, we expect to continue to receive the LRRP, both in draft and final form, and we expect to utilize it in our review of and report on the NRC Safety Research Program and Budget for the Commission and the Congress.

We would be pleased to have your comments on this proposed change in procedures.

Sincerely,

P. Shewmon

P. Shewmon
Chairman

ATTACHMENT 4

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

July 15, 1987

The Honorable Lando W. Zech, Jr.
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Zech:

SUBJECT: ACRS COMMENTS ON THE INTEGRATED SAFETY ASSESSMENT PROGRAM

During the 327th meeting of the ACRS, July 9-11, 1987, we reviewed the experience gained with the NRC Staff's Integrated Safety Assessment Program (ISAP) pilot program. This topic was discussed during a Subcommittee meeting on July 7, 1987. The ISAP process had also been discussed during our 279th meeting in July 1983, as reported in our letter to Chairman Palladino of July 12, 1983. In our review, we had the benefit of discussions with the NRC Staff and industry representatives and of the documents referenced.

The ISAP is intended as a cooperative program, between the NRC Staff and a licensee, which provides for the optimized resolution of multiple safety issues. The program would permit a licensee to develop an integrated plan for implementing plant improvements in response to outstanding safety issues. It would be based on a comprehensive list of issues, an assessment of the plant's operating experience, and a plant-specific risk analysis. The ISAP concept grew out of the Systematic Evaluation Program (SEP), which was highly successful in upgrading the ten oldest licensed plants to conformance with current regulatory requirements.

The ISAP pilot program was carried out for Millstone Unit 1 and Haddam Neck by Northeast Nuclear Energy Company (NNECO) and the NRC Staff. The NRC Staff has issued a draft Integrated Safety Assessment Report (ISAR) for Millstone Unit 1 and will soon issue the draft ISAR for Haddam Neck. The NRC Staff is preparing recommendations to the Commission for the future use of the ISAP process based on an analysis of the lessons learned from the pilot program.

Experience with the ISAP pilot program demonstrates that the process is useful and provides an excellent mechanism for the implementation of regulatory requirements. The NRC Staff and NNECO appear to agree on the usefulness of ISAP. We recommend that the ISAP process be extended to other plants.

The NRC Staff has not yet completed the development of the procedures by which the ISAP process would be implemented more broadly. The methods used in the pilot program were successful. However, several points need to be considered further:

(1) the application of the process to plants that were not in the SEP

(2) the coordination with the Individual Plant Examination program under the Severe Accident Policy implementation

(3) the appropriate scope for the risk analysis to be used as part of ISAP

We would like to review the NRC Staff's proposal for future use of the ISAP when it is available.

Sincerely,

W Kerr

William Kerr
Chairman

References:

1. U.S. Nuclear Regulatory Commission, NUREG-1184, "Integrated Safety Assessment Report," Draft Report, April 1987
2. Northeast Utilities, "Integrated Safety Assessment Program, Haddam Neck Plant," Volumes 1 to 3, Final Report, December 1986
3. Northeast Utilities, "Integrated Safety Assessment Program, Millstone, Unit 1," Volumes 1 to 3, Final Report, July 1986
4. Northeast Utilities Service Company, NUSCO 149, "Connecticut Yankee Probabilistic Safety Study," Volumes 1 to 4, February, 1986
5. Northeast Utilities Service Company, NUSCO 147, "Millstone, Unit 1, Probabilistic Safety Study," Volumes 1 to 3, July 1985

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

March 9, 1987

Mr. Victor Stello
Executive Director for Operations
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Stello:

SUBJECT: ACRS COMMENTS ON THE ADVANCE NOTICE OF PROPOSED RULEMAKING ON THE DEFINITION OF "HIGH-LEVEL RADIOACTIVE WASTE"

During the 323rd meeting of the ACRS, March 5-7, 1987, a discussion was held with the NRC Staff relative to the Advanced Notice of Proposed Rulemaking on the Definition of "High-Level Radioactive Waste." This topic was also discussed during a meeting of the ACRS Subcommittee on Waste Management on February 19 and 20, 1987.

On the basis of this review, we believe that the approach being taken by the NRC Staff is reasonable. We support the Staff's efforts to base the definition of high-level radioactive waste on the associated risks; it should not be based on the source of the waste. Such an approach will provide for better protection of the public as well as better allocation of resources.

We plan to offer additional comments on this matter after responses from the public have been received and evaluated by the NRC Staff.

Sincerely,

W Kerr

William Kerr
Chairman

Reference:
1. Draft Advance Notice of Proposed Rulemaking, 10 CFR Part 60 Definition of "High-Level Radioactive Waste" (undated -- received about February 1, 1987)

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

March 9, 1987

Mr. Victor Stello
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Stello:

SUBJECT: ACRS COMMENTS ON "STANDARD FORMAT AND CONTENT" (NUREG-1199) AND "STANDARD REVIEW PLAN" (NUREG-1200), GUIDANCE DOCUMENTS FOR THE PREPARATION OF A LICENSE APPLICATION FOR A LOW-LEVEL WASTE DISPOSAL FACILITY

During the 323rd meeting of the ACRS, March 5-7, 1987, we met with the NRC Staff to discuss "Standard Format and Content" (NUREG-1199) and "Standard Review Plan" (NUREG-1200), guidance documents for the preparation of a license application for a low-level waste disposal facility. These documents were also discussed during a meeting of the ACRS Subcommittee on Waste Management on February 19 and 20, 1987. On the basis of this review, we offer the following comments.

In general, we conclude that both of these documents are overly detailed and stringent. Both require applicants to submit information and to develop capabilities that do not appear to be warranted by the public health risks associated with a low-level radioactive waste disposal facility.

While too detailed in some respects, the reports are not clear enough in others (for example, in the definition of a "buffer zone"). They contain requirements that may exceed current technical capabilities (such as the verification of the class of a given waste sample, and a determination of whether it contains hazardous toxic chemicals). They also contain discussions of certain topics (such as environmental monitoring) that are so dispersed throughout the reports that they are difficult to follow. Compounding these problems is the fact that, while these two reports cite International Commission on Radiological Protection (ICRP) Publication 30 as the basis for associated radiation dose assessments, the referenced NRC regulations. 10 CFR Part 61, are based on ICRP Publication 2 and the standards for radiation protection as prescribed in 10 CFR Part 20.

We recommend that the NRC Staff simplify and clarify these two documents. It may be useful in this effort for the NRC Staff to review any comparable U. S. Environmental Protection Agency reports prepared for the review of facilities for the disposal of toxic chemical wastes.

Sincerely,

W Kerr
William Kerr
Chairman

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

April 14, 1987

Honorable Lando W. Zech, Jr.
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Zech:

SUBJECT: ACRS COMMENTS ON PROPOSED NUCLEAR WASTE ADVISORY COMMITTEE

During the 324th meeting of the ACRS, April 9-11, 1987, we briefly discussed the recommendations provided to the Commissioners by the Executive Director for Operations on arranging for outside advice and expertise on high level waste issues (SECY-87-91, dated April 3, 1987).

Since we have serious concerns on certain aspects of this proposal, we would appreciate Commission deferral of a decision on this matter until after we have had an opportunity to meet with the NRC Staff next month and to provide you with our comments.

Sincerely,

WKerr

W. Kerr
Chairman

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

June 9, 1987

The Honorable Lando W. Zech, Jr.
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Zech:

SUBJECT: ACRS COMMENTS ON DISPOSAL OF MIXED WASTE

During the 326th meeting of the ACRS, June 4-6, 1987, we heard a report from the Division of Low-Level Waste Management and Decommissioning on its efforts to develop jointly with the U.S. Environmental Protection Agency (EPA) a definition of, and acceptable methods for regulating the disposal of, "mixed waste." This subject was also discussed by our Waste Management Subcommittee during a meeting on May 18-19, 1987.

Our impression is that NRC Staff members responsible for this effort have, in cooperation with the EPA, made significant progress in resolving the relevant issues (namely, siting guidelines, design standards, and the complexities of dual regulation). We commend them for their efforts.

We have some concerns, however, about the interpretation of the definition of "mixed waste." If a strict interpretation results in a large increase in the wastes classified within this category, this could have a negative impact on the disposal of wastes from many facets of the nuclear industry. Specific questions to be addressed in resolving this issue include the procedures and schedule for licensing facilities where such wastes can be disposed, the role of Agreement States in such activities, and how such wastes are to be handled in the interim.

We concur with the NRC Staff's conclusion that substantial work is still required for the dual issuance of EPA permits and NRC licenses, as well as dual inspection and enforcement activities. We request that the NRC Staff keep us informed as progress is made in this area.

Sincerely,

W Kerr

William Kerr
Chairman

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

June 10, 1987

The Honorable Lando W. Zech, Jr.
Chairman
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Zech:

SUBJECT: ACRS COMMENTS ON QUALITY ASSURANCE PROGRAMS FOR A HIGH-LEVEL WASTE REPOSITORY

During the 326th meeting of the ACRS, June 4-6, 1987, we heard a report from the Division of High-Level Waste Management on its review of the quality assurance (QA) program being utilized by the U.S. Department of Energy (DOE) in the development of a high-level waste (HLW) geologic repository. This subject was also discussed during a meeting of our Waste Management Subcommittee on May 18-19, 1987.

We are pleased that the NRC Staff is generally familiar with the QA lessons learned from experience in the nuclear power plant arena as described in the report of the so-called Ford Amendment Study, NUREG-1055 (Reference 1). However, we are concerned that some prior mistakes might be repeated in the HLW repository development process.

In particular, we believe that confirmation of quality should be carried out in discrete steps throughout the lengthy HLW repository development process (e.g., through readiness reviews), in contrast to confirmation near the end of the process only. Although they are not required to do so by regulations, we urge that the DOE and the NRC jointly define acceptance criteria and a schedule for conducting such readiness reviews. Specific hold points should be jointly agreed to in the HLW repository development process. Reviews should be conducted at these hold points, using the acceptance criteria agreed to in advance, to verify quality in the development process up to that point. Confirmation of quality sufficient for licensing would indicate that the development process is ready to proceed toward the next quality review hold point. Failure to confirm the quality of the development process as it proceeds could result in considerable difficulty in licensing the completed repository.

Although the NRC Staff has in the past expressed serious concerns about certain aspects of the DOE quality assurance program, we were pleased to hear a more favorable current report. The NRC Staff should continue monitoring DOE's activities in this area. Of special interest to us is the so-called Q-list (Reference 2) being developed by the NRC Staff for application to

DOE's program. We look forward to receiving this document as soon as it is available in final form.

Concurrent with the above, we encourage the NRC Staff to move forward rapidly in the development of an NRC quality assurance program for application to those portio ' of NRC activities that pertain to its independent evaluation and review of the DOE high-level waste program. The NRC quality assurance program should apply, in particular, to the NRC contractors (e.g., the Federally Funded Research and Development Center) involved in this work.

We request that the NRC Staff keep us informed on the progress of both the DOE and the NRC quality assurance programs.

Sincerely,

WKerr

William Kerr
Chairman

References:

1. NUREG-1055, "Improving Quality and the Assurance of Quality in the Design and Construction of Nuclear Power Plants" (A Report to Congress), dated May 1984 (Reprinted March 1987).
2. Draft Generic Technical Positon on Items and Activities in the HLW Geologic Repository Program Subject to 10 CFR Part 60 Quality Assurance Requirements, dated July 1986

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

September 17, 1987

The Honorable Lando W. Zech, Jr.
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Zech:

SUBJECT: ACRS ACTION ON THE PROPOSED FINAL RULE AMENDMENTS TO 10 CFR PART 72, "LICENSING REQUIREMENTS FOR THE INDEPENDENT STORAGE OF SPENT NUCLEAR FUEL AND HIGH-LEVEL RADIOACTIVE WASTE"

During the 329th meeting of the ACRS, September 10-12, 1987, we discussed a report from our Subcommittee on Spent Fuel Storage regarding the proposed Final Rule Amendments to 10 CFR Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste."

As a result of this discussion, we concur with the NRC Staff's proposal to issue the subject Final Rule Amendments. However, we note that operators of equipment and controls at facilities licensed under 10 CFR Part 72 would not be licensed by the NRC but are required by NRC regulation to be certified by the licensee. Because these facilities could contain large quantities of special nuclear material, qualification and certification of operators is appropriate. Although NRC requires certification of operators, we are not aware of any NRC guidance or criteria that would be used to determine the adequacy of the qualifications, training, continuing training, and certification of these operators. We believe that NRC should develop such guidance and criteria, and we would like to review them before the licensing of such facilities.

Sincerely,

WKerr

William Kerr
Chairman

Reference:
Memorandum from Robert Bosnak, Deputy Director, Office of Nuclear Regulatory Research, dated July 14, 1987, with enclosed proposed Final

Rule Amendments to 10 CFR Parts 2, 19, 20, 21, 70, 73, 75, 150 and Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste."

cc: S. J. Chilk, SECY
T. Rehm, EDO
E. Beckjord, RES
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

November 10, 1987

The Honorable Lando W. Zech, Jr.
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Zech:

SUBJECT: ACRS COMMENTS ON RADIOACTIVE WASTE MANAGEMENT RESEARCH AND OTHER ACTIVITIES

During the 331st meeting of the Advisory Committee on Reactor Safeguards, November 5-7, 1987, we discussed several high-level (HLW) and low-level (LLW) radioactive waste management research activities. We had previously discussed these activities with the NRC Staff during our 329th meeting, September 10-12, 1987. These matters were also discussed during meetings of the Waste Management Subcommittee on August 17-19, 1987 and October 15-16, 1987, and during the Subcommittee's field trip to the University of Arizona on July 28, 1987.

The recent changes in organization of the waste management activities of the NRC have provided opportunities for the proper focusing of attention, both by the NRC Staff and the ACRS, on the LLW and the HLW programs. On the basis of our review of these activities, however, we have noted several potential problems that need to be addressed. In this regard, we offer the following comments.

One of our more important observations is that there is a need for the NRC Staff to better define the scientific bases for some of the requirements specified in various Technical Positions and the connection between these requirements and the NRC regulations they are designed to support. In some cases, these requirements appear to have been introduced only for the convenience of Agreement States or the operators of shallow land burial facilities. We believe that this practice should be carefully examined to determine whether it establishes an undesirable precedent and whether such needs by the States could be accommodated by a method other than the exercise of regulatory power.

An example of this problem is the Technical Position on Low-Level Waste Form. This document demonstrates a need by the NRC Staff to define more clearly the connection between the requirements for testing the waste form and the regulations governing its performance. We recommend that the Division of Low-Level Waste Management and Decommissioning (DLLWMD) Staff reexamine the fundamental bases that led to the formulation of the Technical Position and its requirements, and ensure that the test and performance requirements are pertinent to the conditions likely to be found in shallow land burial facilities. For example, leach testing is now being required of the LLW form. The NRC Staff, however, was not able to demonstrate an explicit connection between this requirement and regulatory criteria. The Staff should be directed either to define such a connection or to delete this requirement. Further, they should document and make readily available the analyses that form the bases of performance evaluation and acceptance of the waste form.

The continued aging of U.S. nuclear power plants makes it likely that the volumes of LLW from decontamination and decommissioning activities will increase. We believe that the complexity of the chemistry of such wastes requires that the DLLWMD Staff formulate very clearly the associated problems and the proper approaches for solving them. As a part of this process, the Staff should seek the support of consultants and/or members of the Office of Nuclear Regulatory Research (RES) contractor staffs.

We reviewed several RES programs dealing with the integrity of the HLW repository. While the results of the NRC research may need to be used in an adjudicatory hearing involving the Department of Energy, our review revealed that the NRC data were obtained under a quality assurance (QA) program considerably weaker than that imposed by NRC on DOE. We believe that the Division of High-Level Waste Management should actively review the NRC research programs and their output, and implement such disciplined QA activities as are needed to provide data with credibility comparable to those of DOE.

Finally, the review of the RES programs revealed that only a very modest level of peer review had been employed. Further, we note that the request for proposal for the Federally Funded Research and Development Center (now called the Center for Nuclear Waste Regulatory Analyses) appeared to discourage the contractor from publishing his results in refereed journals, thereby disallowing the usual form of peer review. In addition to encouraging journal publication, we believe that the Staff should implement a careful, focused, and visible peer evaluation of both the quality of the research results and their applicability to regulatory requirements. Such evaluations should be initiated for each program to the extent feasible, should be periodic, and should be designed to provide clear objectives for the management of the research program.

Sincerely,

W Kerr

William Kerr
Chairman

NRC FORM 335 (9-87) NRCM 1102, 3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

SEE INSTRUCTIONS ON THE REVERSE

1. REPORT NUMBER (Assigned by PPMB, DPS, add Vol. No., if any)

NUREG-1125, Volume 9

2. TITLE AND SUBTITLE

A Compilation of Reports of the Advisory Committee on Reactor Safeguards. 1987 Annual

3. LEAVE BLANK

4. DATE REPORT COMPLETED

MONTH	YEAR
January	1988

5. AUTHOR(S)

6. DATE REPORT ISSUED

MONTH	YEAR
April	1988

7. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)

Advisory Committee on Reactor Safeguards
U. S. Nuclear Regulatory Commission
Washington, DC 20555

8. PROJECT/TASK/WORK UNIT NUMBER

9. FIN OR GRANT NUMBER

10. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)

Same as above

11a. TYPE OF REPORT

Compilation

b. PERIOD COVERED (Inclusive dates)

January thru December 1987

12. SUPPLEMENTARY NOTES

13. ABSTRACT (200 words or less)

This compilation contains 47 ACRS reports submitted to the Commission or to the Executive Director for Operations during calendar year 1987. It also includes a report to the Congress on the NRC Safety Research Program for FY 1988. All reports have been made available to the public through the NRC Public Document Room and the U.S. Library of Congress. The reports are divided into two groups: Part 1: ACRS Reports on Project Reviews, and Part 2: ACRS Reports on Generic Subjects. Part 1 contains ACRS reports alphabetized by project name and within project name by chronological order. Part 2 categorizes the reports by the most appropriate generic subject area and within subject area by chronological order.

14. DOCUMENT ANALYSIS – a. KEYWORDS/DESCRIPTORS

Nuclear Reactors
Nuclear Reactor Safety
Reactor Operations
Safety Engineering
Safety Research

b. IDENTIFIERS/OPEN-ENDED TERMS

15. AVAILABILITY STATEMENT

Unlimited

16. SECURITY CLASSIFICATION

(This page) Unclassified

(This report) Unclassified

17. NUMBER OF PAGES

18. PRICE