A Compilation of Reports of

The Advisory Committee on Reactor Safeguards

U.S. Nuclear Regulatory Commission

April 1992

AVAILABILITY NOTICE

Availability of Reference Materials Cited in NRC Publications

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

1. The NRC Public Document Room, 2120 L Street, NW., Lower Level, Washington, DC 20555
2. The Superintendent of Documents, U.S. Government Printing Office, P.O. Box 37082, Washington, DC 20013-7082
3. The National Technical Information Service, Springfield, VA 22161

Although the listing that follows represents the majority of documents cited in NRC publications, it is not intended to be exhaustive.

Referenced documents available for inspection and copying for a fee from the NRC Public Document Room include NRC correspondence and internal NRC memoranda; NRC bulletins, circulars, information notices, inspection and investigation notices; licensee event reports; vendor reports and correspondence; Commission papers; and applicant and licensee documents and correspondence.

The following documents in the NUREG series are available for purchase from the GPO Sales Program: formal NRC staff and contractor reports, NRC-sponsored conference proceedings, international agreement reports, grant publications, and NRC booklets and brochures. Also available are regulatory guides, NRC regulations in the *Code of Federal Regulations*, and *Nuclear Regulatory Commission Issuances*.

Documents available from the National Technical Information Service include NUREG-series reports and technical reports prepared by other Federal agencies and reports prepared by the Atomic Energy Commission, forerunner agency to the Nuclear Regulatory Commission.

Documents available from public and special technical libraries include all open literature items, such as books, journal articles, and transactions. *Federal Register* notices, Federal and State legislation, and congressional reports can usually be obtained from these libraries.

Documents such as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings are available for purchase from the organization sponsoring the publication cited.

Single copies of NRC draft reports are available free, to the extent of supply, upon written request to the Office of Administration, Distribution and Mail Services Section, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

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, for 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.

NUREG-1125
Volume 13

A Compilation of Reports of The Advisory Committee on Reactor Safeguards

1991 Annual

U.S. Nuclear Regulatory Commission

April 1992

ABSTRACT

This compilation contains 41 ACRS reports submitted to the Commission, Executive Director for Operations, or to the Office c Nuclear Regulatory Research, during calendar year 1991. It also includes a report to the Congress on the NRC Safety Research Program. All reports have been made available to the public through the NRC Public Document Room and the U. S. Library of Congress. The reports are divided into two groups: Part 1: ACRS Reports on Project Reviews, and Part 2: ACRS Reports on Generic Subjects. Part 1 contains ACRS reports alphabetized by project name and by chronological order within project name. Part 2 categorizes the reports by the most appropriate generic subject area and by chronological order within subject area.

PREFACE

The enclosed reports represent the recommendations and comments of the U. S. Nuclear Regulatory Commission's Advisory Committee on Reactor Safeguards during calendar year 1991. NUREG-1125 is published annually. Previous issues are as follows:

Volume	Inclusive Dates
1 through 6	September 1957 through December 1984
7	Calendar Year 1985
8	Calendar Year 1986
9	Calendar Year 1987
10	Calendar Year 1988
11	Calendar Year 1989
12	Calendar Year 1990

ACRS MEMBERSHIP (1991)

CHAIRMAN: Mr. David A. Ward, Retired
E. I. du Pont de Nemours & Company
Savannah River Laboratory

VICE CHAIRMAN: Dr. Paul G. Shewmon
Ohio State University

MEMBERS: Mr. James C. Carroll, Retired
Pacific Gas & Electric Company

Dr. Ivan Catton
University of California, Los Angeles

Dr. William Kerr, Retired (Prof. Emeritus)
University of Michigan

Dr. Thomas S. Kress
Oak Ridge National Laboratory

Dr. Harold W. Lewis
University of California, Santa Barbara

Mr. Carlyle Michelson, Retired
Tennessee Valley Authority and
AEOD/US Nuclear Regulatory Commission

Dr. Chester P. Siess, Retired (Prof. Emeritus)
University of Illinois

Dr. J. Ernest Wilkins, Jr.
Clark Atlanta University

Mr. Charles J. Wylie, Retired
Duke Power Company

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

Rules and Regulations (cont'd)

Safety Research

Waste Management

Part 1: ACRS Reports on Project Reviews

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

March 12, 1991

The Honorable Kenneth M. Carr
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Carr:

SUBJECT: RESTART OF THE BROWNS FERRY NUCLEAR PLANT, UNIT 2

During the 371st meeting of the Advisory Committee on Reactor Safeguards, March 7-9, 1991, we reviewed the status of the resolution of the issues relating to the restart of the Browns Ferry Nuclear Plant (BFNP), Unit 2. Our Subcommittee on TVA Plant Licensing and Restart met on March 4-5, 1991, to discuss this matter and toured several areas of the BFNP on the morning of March 4, 1991. We had the benefit of discussions with the NRC staff and TVA representati' ', as well as the documents referenced.

The BFNP consists of three BWR electric generating units, each rated at 1098 MWe. TVA shut down Unit 2 for refueling in September 1984 and shut down Units 1 and 3 in March 1985 because of NRC concerns regarding declining performance at BFNP. All three units have remained shutdown.

On September 17, 1985, the Executive Director for Operations of the NRC issued a letter to the Chairman of the Board of Directors of TVA requesting information on the actions being taken to resolve NRC's concerns about TVA's nuclear program, including the BFNP. In response, TVA submitted to the NRC a Corporate Nuclear Performance Plan. This plan identified the root auses of the problem in TVA's nuclear program and described measures to remedy the problems at the corporate level.

In addition to its Corporate Nuclear Performance Plan, TVA prepared separate plans to address the problems at each of its nuclear plants. The Browns Ferry Nuclear Performance Plan (BFNPP), Rev. 2, describes the problems and the corrective actions to be taken at Browns Ferry. The BFNPP was specifically directed to the restart of Unit 2, although many of the programs associated with Unit 2 restart have applicability to Units 1 and 3.

TVA determined the problems at Browns Ferry to be the result of three primary causes:

(1) Lack of clear assignment of responsibility and authority to managers and their organizations that clearly established accountability for performance.

(2) Insufficient management involvement and control in the work place leading to a failure to adequately establish highest quality o' performance.

(3) Failure to maintain consistently a documented design basis for the plant and to control consistently the plant's configuration with that basis.

The BFNPP identified specific functional areas of plant activities that were determined to require strengthening on a long-term continuing basis. These included operations, maintenance, surveillance, radiological controls, chemistry, security, emergency preparedness, and scheduling of activities at the site.

During our review, we considered the organizational changes, plant and equipment modifications, and quality control measures that are being implemented to accomplish improvements and corrective actions. We also considered matters related to corporate and plant personnel and personnel training programs. In addition, we were informed of measures that TVA has taken to learn from the nuclear industry, including visits to plants with good operating performance.

During the tour of BFNP, members of our subcommittee observed results of TVA management efforts to improve the working environment and morale of plant employees, to encourage responsiveness, and to establish better lines of communication with employees.

We conclude that the problems and deficiencies that led to the shutdown of BFNP are being addressed adequately. We believe that after TVA has appropriately implemented its commitments and corrective action plans described in the BFNPP to the satisfaction of the NRC staff, the Browns Ferry Nuclear Plant, Unit 2, can be operated without undue risk to the health and safety of the public.

Sincerely,

David A. Ward

David A. Ward
Chairman

References:
1. U.S. Nuclear Regulatory Commission, NUREG-1232, Volume 3, April 1989, "Safety Evaluation Report on Tennessee Valley Authority, Browns Ferry Nuclear Performance Plan;" Supplement 1, October 1989; and Supplement 2, January 1991.
2. Tennessee Valley Authority, Corporate Nuclear Performance Plan, Volume 1, Rev. 6, May 5, 1989.
3. Tennessee Valley Authority, Browns Ferry Nuclear Performance Plan, Volume 3, Rev. 2, October 24, 1988

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

October 18, 1991

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

Dear Chairman Selin:

SUBJECT: DIABLO CANYON NUCLEAR POWER PLANT LONG TERM SEISMIC PROGRAM

During the 378th meeting of the Advisory Committee on Reactor Safeguards, October 10-12, 1991, we reviewed the NRC staff's evaluation of the Long Term Seismic Program (LTSP) carried out by the Pacific Gas and Electric Company (licensee) in connection with its Diablo Canyon Nuclear Power Plant, Units 1 and 2. This evaluation is included in Supplement No. 34 to NUREG-0675, the staff's Safety Evaluation Report for the operation of these plants. The background for the LTSP is described below.

The ACRS provided reports on construction permit applications for Diablo Canyon, Unit 1, in December 1967, and for Unit 2 in October 1969. In both instances, no particular concern was expressed about the seismic design basis, which was 0.2g for the Design Earthquake and 0.4g for what was called the Double Design Earthquake.

In 1971, the Hosgri Fault was discovered and the seismic design bases were reviewed and revised over the next few years. During this period, the ACRS and its consultants in the areas of geology, seismology, and earthquake engineering were involved to a significant extent in the efforts of the staff and the licensee to arrive at new seismic design bases. During this period, the ACRS held ten subcommittee meetings, seven of which related to seismic matters. Three of these seven meetings were held in San Luis Obispo, California, near the site; two in Los Angeles, California; and two in Washington, D.C.

The ACRS review of the operating license application for Diablo Canyon was completed with two subcommittee meetings and a meeting of the full ACRS in June and July 1978. The ACRS report endorsing an operating license was issued on July 14, 1978. This report included extensive discussion of the revised seismic design bases for the plant and reasons for finding them acceptable, and concluded with the following statement:

"The ACRS notes that, for distances less than 10 km from the earthquake source, there are currently no strong motion data for shocks larger than magnitude 6 and few reliable data for shocks of magnitude 5 and 6. Also, the theory and analyses of earthquake and seismic wave generation, of seismic wave transmission and attenuation, and of soil-structure interaction are in a state of active development. The Committee recommends that the seismic design of Diablo Canyon be reevaluated in about ten years taking into account applicable new information."

As a result of this recommendation by the ACRS, the NRC included in the operating license for Diablo Canyon a license condition requiring what became known as the Long Term Seismic Program. The Committee reviewed this license condition at subcommittee and full committee meetings in May and June 1984, and indicated its greement in a report dated June 20, 1984. The operating license was issued in November of that same year.

The licensee and the NRC staff spent the next year developing and reviewing a plan for the conduct of the LTSP. The ACRS reviewed the proposed plan and indicated its agreement in a report dated July 17, 1985. The LTSP was begun in July 1985 and completed in July 1988 -- three years as required by the license condition. During that period, the Committee reviewed progress on the program at subcommittee meetings in November 1986 and February 1988. In addition, the Committee's consultants in the areas of geology and seismology attended numerous meetings at which the results from the program were presented and discussed by the licensee, the NRC staff, and other interested and knowledgeable persons.

The staff's Safety Evaluation Report covering the LTSP was issued in June 1991, after a substantial period of review of the licensee's report and requests for, and submittal of, additional information. Our final review involved a subcommittee meeting in San Luis Obispo on September 16-17, 1991, and review by the full ACRS during its 378th meeting.

At our subcommittee meeting on September 16, 1991, several members of the public expressed the view that the United States Geological Survey (USGS) should be retained by the NRC to perform an independent seismic study of the Diablo Canyon ea. We see no need for such a study. The USGS was retained by the staff as a consultant on geologic and seismologic matters, as were other competent consultants. During progress in the program and in our review of the final report and safety evaluation, we, with the help of our consultants in these areas, have given special attention to the activities of the licensee and the staff relating to geology and seismology. We are satisfied that these programs have been carried out in a competent and professional manner. Those geologic and seismologic characteristics of the area that are significant to the seismic safety of the plant are not at issue among the large number of experts and consultants associated with the licensee, the staff, and the ACRS.

We agree with the staff's conclusion that, subject to resolution of some minor confirmatory items, the License Condition has been met. We believe further that the seismic margins for the plant are adequate and quite comparable to those for other plants in the United States. The results of the probabilistic risk assessment show no significant seismic vulnerabilities. We continue to believe that the Diablo Canyon Nuclear Power Plant can be operated without undue risk to the health and safety of the public.

Mr. James C. Carroll did not participate in the Committee's deliberations regarding this matter.

Sincerely,

David A. Ward

David A. Ward
Chairman

References:
1. U.S. Nuclear Regulatory Commission, NUREG-0675, "Safety Evaluation Report Related to the Operation of Diablo Canyon Nuclear Power Plant, Units 1 and 2, Supplement 34," June 1991
2. Pacific Gas and Electri Company, "Final Report of the Diablo Canyon Long Term Seismic Program," July 1988, and addenda through May 29, 1991

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

March 12, 1991

The Honorable Kenneth M. Carr
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Carr:

SUBJECT: FULL-TERM OPERATING LICENSE FOR THE OYSTER CREEK NUCLEAR GENERATING STATION

During the 371st meeting of the Advisory Committee on Reactor Safeguards, March 7-9, 1991, we reviewed the application by the GPU Nuclear Corporation and Jersey Central Power & Light Company (licensees) for conversion of the provisional operating license (POL) for the Oyster Creek Nuclear Generating Station to a full-term operating license (FTOL). During our review, we had the benefit of discussions with representatives of the licensees and the NRC staff. We also had the benefit of the documents referenced.

The Committee most recently reported on the Oyster Creek Station in a letter dated November 9, 1982, relating to the Systematic Evaluation Program (SEP) review of this plant.

A POL for Oyster Creek was issued in April 1969 and commercial operation began in December 1969. The application for an FTOL was filed in March 1972, but review of this application was deferred by the NRC staff in 1975, along with several other FTOL reviews. In 1978, the Oyster Creek Station was included in Phase II of the SEP because much of the review needed for the FTOL was similar in scope to that for the SEP.

The Committee, in its November 9, 1982 report on the results of the SEP as applied to the Oyster Creek Station, indicated that its review of the FTOL application would be deferred until the NRC staff had completed its actions on the SEP issues that were still pending, and on the Unresolved Safety Issues (USIs) and TMI Action Plan items. All but parts of six of the SEP issues have been resolved to the satisfaction of the NRC staff as reported in Supplement 1 to the Integrated Plant Safety Assessment Report for Oyster Creek. The staff has discussed the status of these six issues and of the USIs and TMI Action Plan items in its Safety Evaluation Report related to the FTOL for Oyster Creek. We believe that the procedures and schedules that have been agreed to for resolving these items are satisfactory and that the remaining actions to resolve these items would not be accelerated by withholding an FTOL at this time.

We believe that there is reasonable assurance that the Oyster Creek Nuclear Generating Station can continue to be operated at power levels up to 1930 MWt under a full-term operating license without undue risk to the health and safety of the public.

Sincerely,

David A. Ward
Chairman

References:

1. U.S. Nuclear Regulatory Commission, NUREG-1382, "Safety Evaluation Report Related to the Full-Term Operating License for Oyster Creek Nuclear Generating Station," January 1991.
2. U.S. Nuclear Regulatory Commission, NUREG-0822, Supplement No. 1, "Integrated Plant Safety Assessment, Systematic Evaluation Program, Oyster Creek Nuclear Generating Station," July 1988.
3. Letter dated February 14, 1991, from James M. Taylor, Executive Director for Operations, NRC, to Philip R. Clark, President, General Public Utilities Nuclear Corporation, forwarding Diagnostic Evaluation Team Report for Oyster Creek Nuclear Generating Station.

August 13, 1991

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

Dear Chairman Selin:

SUBJECT: FULL-TERM OPERATING LICENSE FOR THE SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 1

During the 376th meeting of the Advisory Committee on Reactor Safeguards, August 8-9, 1991, we completed our review of the application by the Southern California Edison Company and San Diego Gas and Electric Company (licensees) for conversion of the provisional operating license (POL) for the San Onofre Nuclear Generating Station, Unit 1, to a full-term operating license (FTOL). During our review, we had the benefit of discussions with representatives of the licensees and the NRC staff. We also had the benefit of the documents referenced. The Committee most recently discussed and reported on this plant in a letter dated August 13, 1985, relating to the Systematic Evaluation Program (SEP) review of San Onofre, Unit 1.

San Onofre, Unit 1, received a POL in March 1967 and began commercial operation in January 1968. The licensees applied for an FTOL in July 1970, but review of this application was deferred by the NRC staff in 1975, along with several other FTOL reviews. In 1978, San Onofre, Unit 1, was included in Phase II of the SEP because much of the review needed for the FTOL was similar in scope to that for the SEP.

The Committee, in its August 13, 1985 letter reporting on the results of the SEP as applied to San Onofre, Unit 1, indicated that its review of the FTOL would be deferred until the NRC staff had completed its actions on the SEP issues that were still pending, and on the Unresolved Safety Issues and TMI Action Plan items. The status of outstanding issues in all of these categories has been discussed by the staff in its Safety Evaluation Report related to the FTOL for San Onofre, Unit 1 (NUREG-1443), and a schedule for their resolution has been established by the licensees and confirmed by the staff in its confirmatory order dated January 2, 1990. We believe that the procedures and schedules that have been agreed to for the resolution of these items are satisfactory, and that the remaining actions to resolve these items would not be accelerated by withholding an FTOL at this time.

We believe that there is reasonable assurance that the San Onofre Nuclear Generating Station, Unit 1, can continue to be operated at power levels up to 1347 MWt under a full-term operating license without undue risk to the health and safety of the public.

Sincerely,

David A. Ward
Chairman

References:

1. U.S. Nuclear Regulatory Commission, NUREG-1443, "Safety Evaluation Report Related to the Full-Term Operating License for San Onofre Nuclear Generating Station, Unit 1," July 1991.
2. U. S. Nuclear Regulatory Commission, NUREG-0829, "Integrated Plant Safety Assessment, Systematic Evaluation Program, San Onofre Nuclear Generating Station, Unit 1," December 1986.

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

February 12, 1991

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

Dear Mr. Taylor:

SUBJECT: EDO RESPONSE TO ACRS REPORT DATED DECEMBER 12, 1990 ON THE PRELIMINARY DESIGN APPROVAL FOR THE RESAR SP/90 DESIGN

In our December 12, 1990 report to Chairman Carr regarding Westinghouse's Application for Preliminary Design Approval for the RESAR SP/90 Design, we expressed a concern (item 4.1) about the location of the emergency diesel generator (EDG) on the same floor and corridor as the control room.

In our report we stated that, "We believe that another location for the EDG room should be specified in view of the potential for fire and/or explosions associated with the operation of large diesel generators."

Item 8 of the enclosure to your response of January 14, 1991 states that, "The staff has not in the past considered, and does not now consider, credible an explosion in the EDG room of sufficient size to cause catastrophic failure of the reinforced concrete enclosure of these rooms."

Your response did not address the large door that separates the EDG from the corridor leading to the control room. We ask that you expand your reply to include consideration of this door and give us your views on the size of a fire and/or explosion that you would consider credible, and some estimate of the structural capability of this door under differential pressure conditions. Also, we ask that you address the potential for a fire resulting from combustibles such as fuel oil that may flow under the door into the corridor.

Sincerely,

David A. Ward

David A. Ward
Chairman

Reference:
Letter dated January 14, 1991, from James M. Taylor, Executive Director for Operations, to David Ward, Chairman, Advisory Committee on Reactor Safeguards, Subject: Report by the Advisory Committee on Reactor Safeguards (ACRS) on the RESAR SP/90, December 12, 1990.

2

Part 2: ACRS Reports on Generic Subjects

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

July 18, 1991

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

Dear Chairman Selin:

SUBJECT: SCHEDULES FOR ADVANCED REACTOR REVIEWS

During the 375th meeting of the Advisory Committee on Reactor Safeguards, July 11-13, 1991, we discussed the staff's proposed "realistic" schedules identified in SECY-91-161 for completing the reviews of the evolutionary and passive advanced light water reactor (ALWR) design certification applications and the review of the Electric Power Research Institute's (EPRI) ALWR Utility Requirements Document. We had the benefit of presentations by and discussions with members of the NRC staff and NUMARC, as well as the documents referenced. Consideration of this matter by the Committee was based on the request of the Commission, as reflected in Staff Requirements Memorandum M910607A dated June 18, 1991.

We believe that, barring unforeseen circumstances, the ACRS will be able to meet these schedules. Note, however, that the time required for Committee review of the final SERs and FDAs will be three months, as stated in the text of SECY-91-161, rather than two months as shown on the bar charts.

Sincerely,

David A. Ward

David A. Ward
Chairman

References:
1. U.S. Nuclear Regulatory Commission, SECY-91-161, dated May 31, 1991, from J. Taylor, Executive Director for Operations, for the Commissioners, Subject: Schedules for the Advanced Reactor Reviews and Regulatory Guidance Revisions
2. Electric Power Research Institute, Utility Requirements Document, June 1986
3. Memorandum dated June 18, 1991 from Samuel J. Chilk, Secretary of the Commission, for David A. Ward, ACRS, and James M. Taylor, EDO, Subject: Staff Requirements - Periodic Meeting with the ACRS, June 7, 1991

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

August 13, 1991

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

Dear Chairman Selin:

SUBJECT: ADDITIONAL COMMENT ON SCHEDULES FOR ADVANCED REACTOR REVIEWS

In our report to you of July 18, 1991, on "Schedules for Advanced Reactor Reviews," we noted that the time required for Committee review of the final Safety Evaluation Reports (SERs) and Final Design Approvals will be three months, as stated in the text of SECY-91-161, rather than two months as shown on the bar charts. We failed to note that the three months review time (starting at time of receipt) also applies to the draft SERs. Except for ABWR, the bar charts show only one mo h for ACRS review. The text is silent on this point.

Sincerely,

David A. Ward
Chairman

<u>Reference</u>:

U.S. Nuclear Regulatory Commission, SECY-91-161, dated May 31, 1991, from J. Taylor, Executive Director for Operations, for the Commissioners, Subject: Schedules for the Advanced Reactor Reviews and Regulatory Guidance Revisions

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

February 12, 1991

The Honorable Kenneth M. Carr
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Carr:

SUBJECT: PROPOSED SUPPLEMENT 4 TO GENERIC LETTER 88-20, INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS (IPEEE) FOR SEVERE ACCIDENT VULNERABILITIES - 10 CFR 50.54(f)

During the 370th meeting of the Advisory Committee on Reactor Safeguards, February 7-9, 1991, we reviewed the NRC staff's resolution of public comments on, and the resulting changes to, the proposed supplement to Generic Letter 88-20, Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities. During this review, we had the benefit of discussions with representatives of the NRC staff. We also had the benefit of the document referenced.

In our May 15, 1990 report to you concerning this subject, we asked for an opportunity to review the final draft of the proposed supplement after the public workshop and resolution of any comments. We have completed this review and conclude that the changes resulting from the resolution of comments are acceptable. Based on our further discussions during this meeting, we have identified the following concerns that we believe should be resolved before the supplement is issued.

1. The staff is asking the licensee to identify vulnerabilities that are discovered in the course of the Individual Plant Examinations (IPEs). In its June 9, 1987 report to Chairman Zech on IPE guidance, the ACRS pointed out that, "Vulnerabilities are not defined, either qualitatively or quantitatively. ., nor is there guidance as to the amount and kind of improvement that the NRC staff will find acceptable." We still find that the staff has not provided either a definition of a vulnerability or guidance on how to identify one, nor does it plan to do so. The staff does plan to review the licensee's IPE, and we were told that if vulnerabilities not identified by the licensee are discovered, the licensee will be asked and, if necessary, required to deal with them. However, even at the review stage, the staff will not provide guidance as to what constitutes a vulnerability.

We believe it would contribute to a more disciplined review, and would provide helpful guidance to licensees if the staff provided, at the very least, some indication of the process to be used in identifying a vulnerability.

2. The staff has had to cope with the problem posed by the existence of two widely different, but equally authoritative, seismic hazard curves, traditionally called the EPRI and the LLNL curves. The staff position is that a licensee may conduct its seismic analysis separately using each of the curves, but may alternatively choose to use only one, provided the one chosen is the "more conservative" of the two. The justification provided was that this procedure is more likely to identify all the relevant accident sequences.

   The use of the word "conservative" may be a problem. The difference between the two curves has nothing to do with increased conservatism but simply reflects two different, and apparently equally valid, technical approaches. Further, conservatism should play no role in an analysis intended to uncover the vulnerabilities of a plant. If there is no technical basis for choosing one hazard curve over the other, the statistically valid procedure is to take a suitable average.

In our report of May 15, 1990, we stated that a simplified fire risk evaluation method is being developed by NUMARC, but has not yet been evaluated by the staff or by us. We are still planning to review the NUMARC method.

Sincerely,

David A. Ward

David A. Ward
Chairman

Reference:
Memorandum dated January 11, 1991 from Warren Minners, Office of Nuclear Regulatory Research, to Raymond F. Fraley, ACRS, Subject: ACRS Review of Individual Plant Examination for Severe Accident Vulnerabilities Due to External Events (IPEEE) - 10 CFR 50.54(f) (Generic Letter 88-20, Supplement 4), with enclosures (Predecisional)

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

May 17, 1991

The Honorable Kenneth M. Carr
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Carr:

SUBJECT: PROPOSED CRITERIA TO ACCOMMODATE SEVERE ACCIDENTS IN CONTAINMENT DESIGN

During the 373rd meeting of the Advisory Committee on Reactor Safeguards, May 8-11, 1991, we discussed development of criteria that would incorporate explicit consideration of severe accidents into requirements for containment design. This matter was also considered during our meetings in December 1990 and in January, February, March, and April 1991. The Committee had also discussed this matter in a number of previous meetings, including discussions with the Commission, the latest on November 8, 1990. In addition, we have had the benefit of discussions with a large number of experts on containment and severe accidents, including representatives from industry, private consultants, the NRC staff, and national laboratories, in a series of ACRS joint Containment Systems and Structural Engineering Subcommittee meetings over the past three years. The Commission had earlier requested an ACRS study of this matter (see Staff Requirements Memorandum of July 28, 1988) based on discussions during an ACRS meeting with the Commission on July 14, 1988.

Our purpose in writing this report is to describe and recommend a possible course by which the NRC could develop an improved set of requirements for the design of containment systems for future nuclear power plants. These requirements would include definition of specific challenges posed by severe accidents. They would be promulgated by revisions and additions to 10 CFR Part 50, primarily to Appendix A, "General Design Criteria for Nuclear Power Plants." Implementation also would require new regulatory guides (RGs). More detail about rule changes and regulatory guides is provided in the Appendix.

We intend this to be a description of a general approach that could be taken. Guidelines for the regulatory guides are provided primarily to illustrate that approach. Final detail and quantification should be developed and justified by the staff with input and review by industry and the reactor safety community.

The new requirements would be applicable to future plants, those not yet designed. We would exclude the "evolutionary" LWRs, for which designs are well advanced. We believe the new criteria can and should be adopted for use in the development and licensing of the "passive" plant designs.

An alternative or interim approach would be to adopt the general process we propose as an extension of the "Policy Statement on Severe Accidents Regarding Future Designs and Existing Plants" published in August 1985. This could be more easily and rapidly adopted, in comparison with the rulemaking approach, as a guide for designers and staff reviewers, and as a basis for design certification. A disadvantage is that the "policy" approach would be subject to less rigorous reviews and more limited input from the general body of available expertise on severe accidents and containment performance. We recommend the rulemaking approach.

Future licensing responsibilities of the NRC may include nuclear power plants other than LWRs. Our proposal is for application only to LWRs. As discussed above, we propose that new containment requirements be implemented through changes in appropriate sections of the General Design Criteria. The introduction to 10 CFR Part 50, Appendix A (issued in 1971), states that these criteria apply for "water-cooled nuclear power plants similar in design and location to plants for which construction permits have been issued by the Commission." There are some general principles that could be applicable to other types of plants. Such application is, however, a task for another day.

There will be debate over our proposal. It will center on the question of whether it is better to continue with the present set of requirements, which it might be argued are good enough, or to develop requirements that reflect what has been learned about severe accidents over the past decade. A classical conflict between short-term and long-term costs and benefits exists. We recommend that development of new containment design criteria proceed along the lines we have proposed. We believe that benefits in safer and more efficiently designed plants and in stabilization of an important part of the regulatory process will be substantial. We look forward to the opportunity to interact with you and the staff on this important subject.

Sincerely,

David A. Ward

David A. Ward
Chairman

APPENDIX

Proposed Criteria to Accommodate Severe Accidents in Containment Design

Table of Contents

I. BACKGROUND

The primary purpose of the containment and its associated systems in an LWR plant is to mitigate the consequences of severe accidents, those which involve fuel melting and an abundant release of fission products. Other important purposes of the containment include: housing the nuclear steam supply system and protecting it from external threats, shielding the environment from radiation emanating from the reactor system, and mitigating the releases of radioactive substances caused by normal operation or incidents of lesser scope than severe accidents.

Although this primary purpose has been recognized from the beginning, and is perhaps obvious, existing NRC requirements do not account for many severe accident phenomena that could challenge a containment's ability to perform its function.

In the early 1960s, licensing authorities and the reactor safety community (including the ACRS) recognized that the risk of a severe accident was real, but remote and largely undefined. Rather than await the results of what was seen to be a long and difficult research effort to understand more about severe accidents, a decision was made to use a surrogate accident as a design basis for the containment and to move forward with the development of nuclear power. That surrogate accident, a sudden large-break LOCA, coupled with the siting criteria in 10 CFR Part 100, has been the basis for LWR containment design ever since.

During the 1979 accident at Three Mile Island 2, a containment designed to the surrogate requirements functioned effectively to protect the public. On the other hand, severe accident research and risk assessments performed since 1979 indicate that a broad range of high-energy loads and fission product releases, more severe than at Three Mile Island 2, might threaten containment systems. There are indications that certain unlikely severe accident challenges could cause containments to fail, and lead to the release of health-threatening quantities of fission products. While the predicted risk from those accidents is small, uncertainty in quantification of the risk is large. Improvements in the design of containments could reduce both the risk and the uncertainty.

II. REVIEW OF EXISTING CONTAINMENT REQUIREMENTS

Formal criteria by which acceptable reactor containments were to be designed and built were established by the Atomic Energy Commission in the 1960s and 1970s. General Design Criteria for water-cooled nuclear power plants (10 CFR Part 50,

Appendix A), promulgated in 1971, included the following requirements relating to containment:

- Criterion 16 specifies "an essentially leak-tight barrier" between the reactor systems and the environment as one of "multiple fission product barriers."

- Criteria 38 through 40 require systems to remove decay heat from the containment to negate pressure buildup that would otherwise result.

- Criteria 41 through 43 provide for a system to remove fission products from the containment atmosphere to reduce the consequences of ongoing leakage.

- Criterion 50 requires that the containment structure be able to accommodate "the calculated pressure and temperature conditions resulting from any loss-of-coolant accident." This is to be accomplished "without exceeding a design leakage rate and with sufficient margin." It states that the margin should reflect consideration of (1) potential energy sources such as energy in steam generators, limited metal-water reaction that might result from degradation but not failure of the ECCS, (2) limited information on accident phenomena, and (3) conservatism in the calculations. There is no requirement in GDC 50 to accommodate severe accidents. However, this was remedied in part by 10 CFR 50.34(f) for near term operating licenses, 10 CFR 52.47 for standard design certification, and 10 CFR 50.44 for combustible gas control.

- Criteria 51 through 57 provide requirements for containment materials, testing, penetrations and isolation.

Reactor siting criteria in 10 CFR Part 100, established in 1962, indirectly determine the maximum leakage rate for which the containment is to be designed. Section 100.11 establishes dose limits for the whole body and for the thyroid. A referenced document, TID-14844, suggests amounts of radioactive material within containment that are to be assumed in calculating hypothetical doses from post-accident containment leakage. TID-14844 also suggests a leakage rate of 0.1 percent of the containment volume per day.

Additional guidance is provided in two regulatory guides originally issued in 1970. Regulatory Guide 1.3 is for BWRs and Regulatory Guide 1.4 is for PWRs. Each specifies the proportions of the elemental, particulate, and organic forms of the radioiodines that are to be assumed in making dispersion and dose calculations. These are, respectively, 91 percent, 5 percent, and 4 percent. In addition, Regulatory Guide 1.4 permits the assumption that the leakage rate from containment for PWRs is reduced to one-half the value given in technical specifications after the first 24 hours.

III. WHY NEW CRITERIA ARE NEEDED

A first purpose of new containment requirements will be to reduce the risk and uncertainty by more directly accounting for severe accident threats than is done with present requirements. This should be feasible because in 1991 more is known about the nature of severe accident threats than was known in 1971. Our proposal is simply a way of applying this improved knowledge to provide improved containment systems.

A second purpose is to clarify what is expected of applicants and to bring greater coherence to the design review and certification processes. Many severe accident considerations are now being factored into staff reviews of advanced reactor designs, but, the process by which this is done is not well defined.

A third purpose is to help ensure that containments will have greater "robustness." A containment cleverly and narrowly designed to mitigate a set of accidents that has been precisely identified may not be able to cope with the unexpected. A truly "robust" containment would have improved capability to deal with the unexpected. A containment that has been designed with explicit consideration of a more extensive set of challenges is likely to be more robust than one designed with consideration of only a limited set.

IV. PROPOSED APPROACH TO DEVELOPMENT OF NEW CONTAINMENT DESIGN CRITERIA

We have previously recommended (ACRS report of May 13, 1987 regarding Safety Goal Policy) a conditional 10 percent failure probability for the containment, reflecting our judgment about the need for assurance of containment performance. It is worth recalling that our recommendation was meant as a hedge against uncertainty, to preserve the concept of defense in depth -- itself a hedge against uncertainty. If all calculations were accurate and credible, all that would matter would be that the population of plants meets the Commission's safety goals, and the identification of containment performance as a separate item would be inappropriate. It is because quantitative risk estimates are not perfect that defense in depth is a useful philosophy, and that separate containment performance guidelines make sense.

The containment performance objective should serve as guidance to the NRC staff in judging whether requirements for containment design properly reflect the intent of the Commission as expressed in the Safety Goal Policy. The conditional containment failure probability should not be simply passed on to applicants for plant licenses. Instead, we propose a two-step process to establish new requirements.

First, the General Design Criteria (GDC) in 10 CFR Part 50 would be revised to acknowledge that containments should be designed for a range of challenges that can threaten their function during severe accidents. Several different challenges or containment loads would be defined, as discussed in Section V of this Appendix. For each, the nature of the challenge would be described in general terms; specifics and quantification would be relegated to a regulatory guide. Also, for each, a success criterion would be specified. In most cases, success would be defined simply as maintenance of the containment function for an appropriate period following the particular challenge. In addition to the GDC changes, certain other regulations concerned with containment would be modified. A summary of proposed regulatory changes is given in Table I.

Second, new regulatory guides would be developed to detail acceptable means to implement the design requirements. For the severe accident requirements of GDC 50, regulatory guides would address each challenge.

The regulatory guides would provide technical definitions of acceptable means of meeting the general design criteria for containment. What we have in mind is a relationship between each GDC requirement and its companion regulatory guide similar to the existing relationship between GDC 35, "Emergency Core Cooling," and Appendix K to 10 CFR Part 50, "ECCS Evaluation Models." GDC 35 states that a system shall provide "abundant emergency core cooling." Appendix K gives, in reasonably unambiguous language, a technical definition of the leak that must be accommodated and a definition of the terms "abundant" and "cooling."

Our revised GDC 50 would state the requirement that containments must have the capability to accommodate a specific list of challenges without loss of containment function. For each challenge, a regulatory guide would define in unambiguous technical terms, first the challenge, and second, what is meant by the term "accommodate."

The technical content of each regulatory guide should provide as complete and unambiguous a basis for containment system design as can be practically developed. For example, the criterion for capacity to accommodate hydrogen combustion might state the total amount of hydrogen to be considered, as a percentage of that which could be generated by complete oxidization of cladding in contact with active fuel, and then require a specific analysis for mixing and stratification. The regulatory guide would describe acceptable mixing models, based on containment type.

An important aspect of what we are proposing is that the NRC will take responsibility for the important technical judgments necessary to transform knowledge from severe accident research and risk assessments into criteria and requirements that can be used by a designer. This would not be done in isolation; review and input from the industry and the reactor safety community should be sought as the rule changes and regulatory guides are developed.

In the following sections, we propose revisions to the regulations relating to containment design requirements and also provide information on the content of proposed regulatory guides. Although we have not attempted to couch the GDC proposals in regulatory language, we believe that the scope of our description is close to the appropriate scope for the rule. In contrast, our proposals for regulatory guides are intended to be only the bare bones of what the guides should contain. It will be up to the staff to develop quantifications and to provide appropriate justification. We will want to interact with the staff as final details are developed.

V. RECOMMENDED CHANGES AND ADDITIONS TO 10 CFR PART 50, APPENDIX A, "GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANTS"

We recommend that the following General Design Criteria be changed as indicated:

Criterion 16 -- Containment Design

This criterion specifies an "essentially leak-tight barrier . . for as long as postulated accident conditions require." No changes in wording are necessary but implications of the words would be different. A regulatory guide would specify a definition for leak-tight that is consistent with the overall package of containment requirements. Existing regulatory guides suggest a leakage rate of 0.1 percent of containment volume per day. Present information about severe accidents and the role of containment suggest leakage of 1 percent may be more appropriate. In addition, the accident conditions for which such a leakage limit would apply should reflect other requirements, in particular those in the new GDC 50.

Criteria 38-40 -- Containment Cooling

These requirements would be changed to reflect the demands placed upon containment cooling systems by other new requirements, especially the proposed new GDC 50(f) and 50(g) below.

Criteria 41-43 -- Containment Atmosphere Cleanup

These requirements would be changed to reflect the demands placed upon containment atmosphere cleanup systems by other new requirements, especially the proposed new GDC 50(f) below.

Criterion 50 -- Containment Design Basis

This criterion would be extensively expanded to require that containment systems be designed to accommodate a variety of challenges that could be created by severe accident conditions. We believe that the challenges can be adequately represented by the eight examples discussed below. Each would be defined in a section [(a) through (h)], with a success criterion identified and with appropriate supporting regulatory guides. These are not meant to be accident scenarios, but are representative phenomenological challenges.

50(a) Loss of Coolant Accident (LOCA)

The containment system would have the capacity to accommodate pressure and temperature conditions resulting from the blowdown of fluid from a large break LOCA; and in the case of PWRs, from a nonconcurrent blowdown of the secondary system.

Leakage should not exceed the rate specified in Criterion 16 for an appropriate period following the accident.

50(b) Fuel-Coolant Interaction

The containment would have capacity to accommodate missiles that could be produced by credible steam explosions within the vessel and to accommodate pressure pulses that could be produced by credible steam explosions outside the reactor vessel and within containment. Steam explosions are characterized by the rapid transfer of thermal energy from molten material to water. Where appropriate, the addition of chemical energy to the thermal energy source would be included in performance calculations.

Leakage should not exceed the rate specified in Criterion 16 following the missile impact or the pressure pulse, crediting dynamic response of the containment structure.

50(c) <u>Hydrogen Combustion and Detonation</u>

The containment would have capacity to accommodate pressure pulses produced by static or shock loadings resulting from the combustion or detonation of hydrogen produced during severe accidents. Hydrogen sources to be considered are the in-vessel and ex-vessel oxidation of core materials, including (1) core degradation from overheating and melting, (2) steam explosions or high pressure melt ejection in the presence of water, and (3) interaction between molten core material and concrete.

Leakage should not exceed the rate specified in Criterion 16 following the pressure pulse, crediting dynamic response of the containment structure.

50(d) <u>Melt Attack on Containment Structure or Pressure Boundary</u>

The containment design would preclude potential for damage to the containment pressure boundary or essential structure by direct contact of molten core material.

Leakage should not exceed the rate specified in Criterion 16 for an appropriate period following the melt attack.

50(e) <u>High Pressure Melt Ejection</u>

The containment system would have the capacity to accommodate rapid increases in static pressure and temperature caused by heating of the containment atmosphere through the direct transfer of thermal and chemical energy from molten core material ejected at high pressure into the containment, unless such ejection is precluded by design of the reactor system.

Leakage should not exceed the rate specified in Criterion 16 for an appropriate period following the melt ejection.

50(f) <u>Corium-Concrete Interaction</u>

The containment system would have the capacity to accommodate the following challenges resulting from the thermal decomposition of concrete by molten corium: (1) the degradation of containment cooling and of cleanup capability due to aerosol formation, (2) slow overpressurization resulting from the evolution of noncondensable gases, (3) functional degradation of structural concrete by erosion, including basemat penetration, and (4) combustion of carbon monoxide.

Challenges to the containment should not be sufficient to render inoperable that equipment required for containment cooling or atmospheric cleanup, nor to cause leakage in excess of the rate specified in Criterion 16 or to allow any release through the basemat within an appropriate time of the onset of the corium-concrete interaction sufficient to cause significant contamination of the groundwater.

50(g) Pressurization from Decay Heat

The containment system would have the capability to accommodate the long-term buildup of pressure resulting from decay heat. This could include an appropriate containment venting system.

Leakage should not exceed the rate specified in Criterion 16 for an appropriate period following the accident.

50(h) Elevated Temperatures

Containment penetrations, equipment necessary for accident management, essential instrumentation, and key structural components would have the capacity to accommodate exposure to elevated containment temperatures.

Exposure of the noted systems and components following exposure to elevated temperatures should not be sufficient to cause leakage in excess of the rate specified in Criterion 16 or damage sufficient to render inoperable that equipment necessary for accident management for an appropriate period following the exposure.

Criteria 51-53

No changes in these criteria are proposed.

Criteria 54-57

These would be revised to be consistent with new Criterion 58. Simplification of Criteria 54-57 may be possible.

In addition to the revisions to existing criteria, described above, we recommend the following new criteria as additions to Appendix A:

Criterion 58 -- Provision For On-Line Monitoring of Containment Isolation Status

This new criterion would be intended to reduce the likelihood of loss of containment function by continuous on-line monitoring. It must be consistent with Criterion 16.

Criterion 59 -- Role of Containment Structure in Protecting Nuclear Components Against External Threats

This new criterion would be intended to protect the nuclear steam supply system and other essential components against credible aircraft crashes, explosions, and other nonnatural threats external to the plant. Alternatively, the existing Criterion 2, which calls for resistance to extreme natural conditions, could be revised to include such threats.

Criterion 59-A - Assurance of Containment Integrity During Shutdown

This new criterion would require that containments will be designed to provide for ease of emergency closure during shutdown operation including station blackout conditions.

VI. RECOMMENDED CHANGES TO OTHER REGULATIONS RELATING TO CONTAINMENT DESIGN

10 CFR Part 100, Reactor Site Criteria

The NRC staff has in progress a study (see SECY-90-341) which would uncouple siting requirements from specifics of plant and containment design. In our report of June 13, 1990, we commented on this program and endorsed the general approach envisioned by the staff.

10 CFR Part 50, Appendix J, Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors

If the allowable leakage rate for accident conditions is increased, and if on-line monitoring capability is provided and used, the requirements of Appendix J would have to be modified extensively. Significant simplification of testing requirements should be possible.

10 CFR 50.34(f) Additional TMI-Related Requirements

Additional requirements pertaining to containment design were promulgated following the TMI-2 accident and are given in 10 CFR 50.34(f). For example, a minimum containment design pressure of 45 psig is specified in one of these. These requirements also apply to standard plant designs to be considered under 10 CFR Part 52. Some of these requirements would be superseded by the expanded GDC 50.

10 CFR 50.44, Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors

Requirements in this section, intended for control of combustible gas generated during severe accidents, should be superseded by new GDC 50(c).

VII. RECOMMENDED CONTENT OF THE NEW REGULATORY GUIDES FOR GENERAL DESIGN CRITERION 50

Implementation of our recommended GDC 50 will require development of new regulatory guides. In what follows are some examples of what these guides should contain. In the final regulatory guides, to be developed by the staff, each should give acceptable values of the important parameters as well as acceptable methods for their calculation. Realistic methods of calculation should be employed. Our recommendations are keyed to the proposed GDC 50 [(a)-(h)].

50(a) Loss-of-Coolant Accident (LOCA)

This regulatory guide should address the current practices for considering LOCAs and indicate the following additions or changes. The best estimate methodology of 10 CFR 50.46 should be used. Active cooling systems should not be credited in calculating maximum containment pressures during blowdown. The effect of thermal stratification on thermal stresses in steel liners and penetrations should be considered.

50(b) Fuel-Coolant Interaction (FCI)

In-Vessel FCI

This regulatory guide should treat PWRs and BWRs separately to account for differences in core degradation processes and differing amounts of zircaloy relative to other materials in the core. For PWRs with safety depressurization systems or with low-pressure sequences of importance, in-vessel FCI should be considered. What constitutes an acceptable mechanistic treatment to establish the quantities of molten core and its temperature should be delineated. Examples of acceptable methods for calculating the mechanical energy produced by FCI should be given. Still further it should specify, for example, that missile velocity be calculated with consideration given the vent path through the downcomer and possible lower head failure. For present-day BWRs, in-vessel FCI is not expected. Future BWRs should be reviewed to be certain this conclusion is still valid.

Ex-Vessel FCI

For FCI outside the vessel, e.g., in a water-filled cavity under the reactor, somewhat different assumptions would be appropriate. Conditions at the time of vessel failure should be used to prescribe the amount and composition of the core material, and its temperature, that need to be considered for evaluation of ex-vessel FCI potential. This regulatory guide should indicate what is acceptable as well as what is an acceptable method for its calculation.

Containment designs that do not preclude water from being in the reactor cavity at the time of vessel failure must consider ex-vessel FCI. This regulatory guide should indicate acceptable methods for calculation of the amount, composition and temperature of the molten core materials at the time of failure, and the type of vessel failure and mass flow rate of molten materials. These calculated values should be used in calculating the mechanical energy produced by the FCI. The mechanical energy calculation is to be based on the same method as described above.

50(c) Hydrogen Combustion and Detonation

There will be different amounts of hydrogen generated by the different reactor types. This regulatory guide should specify the amount of metals oxidized in-vessel and ex-vessel as percentages of what is available, as well as give guidelines as to what constitutes an acceptable mechanistic method for calculating the rate and amount of hydrogen produced. Hydrogen is produced following vessel failure during (1) interactions with water in the cavity, if it exists, and (2) subsequent corium-concrete interaction. This regulatory guide should give guidance as to how much metal is oxidized in each of these two phases of the accident and give guidance to those who wish to calculate it themselves.

Hydrogen in the containment atmosphere can lead to combustion, deflagration, or detonation. All must be considered. To deal with detonation, this regulatory guide should indicate what hydrogen control methods are acceptable and give both acceptable peak pressure and pressure pulse shape with guidance as to how they can be calculated. This guide should also give examples of acceptable analysis methods for calculation of hydrogen distribution within the containment. Pressure calculations should include the effect of hydrogen burns as well as carbon monoxide from corium-concrete interaction with account taken of the timing of the various gas generation processes. The noncondensables from the corium-concrete interaction should also be considered in pressure calculations.

50(d) Melt Attack on Containment Structure or Pressure Boundary

This regulatory guide should contain acceptable values of the molten core material composition, temperature, and rate at which it pours out of the vessel breach, as well as guidelines for an acceptable analysis. Presence of water in the cavity under the reactor should be assumed if the plant is so configured. If justified by a credible spreading analysis, uniform spreading may be assumed. Otherwise, consequences of nonuniform melt depths should be considered. Appropriate heat transfer calculations should be required to establish the thermal insult to the pressure boundary or essential structures.

50(e) High Pressure Melt Ejection

This regulatory guide would apply only to PWRs and only if a depressurization system is not available. It should give guidance on what constitutes acceptable analysis for calculation of thermal energy and corium composition shown to be credible at the time of failure of the reactor pressure vessel. The regulatory guide should indicate that the amount, composition, flowrate, and temperature of the molten material be calculated by an acceptable method. The containment atmosphere should be assumed to be saturated with water vapor. Presence of water in the cavity under the reactor should be included in the analysis if the plant is so configured. Allowable amounts of de-entrainment along the flow path should be specified or methods for their calculation should be given. Oxidation of and heat transfer from the entrained debris should be based on mechanistic modeling.

50(f) Corium-Concrete Interaction

This regulatory guide would be the same for all reactor types. It should specify that a mechanistic evaluation of corium-concrete interaction be performed. The results of an acceptable core melt and vessel failure analysis, defined in this guide, should be used to define the core melt characteristics as it arrives on the reactor cavity floor. Water in the reactor cavity should be accounted for in calculations. The basemat must be shown to be thick enough to provide an appropriate interdiction time before penetration. With consideration given to timing, the contribution of combustibles and noncondensables to containment atmosphere pressure and temperature should be accounted for. Selection of concrete types that reduce gas generation and the use of refractory materials should be encouraged. Core debris control devices and filtered venting for long term pressure control should not be precluded by this guidance.

50(g) Pressurization by Steam From Decay Heat

This regulatory guide should allow for credit to be taken for the decrease in decay heat with time, for heat transfer across the containment boundary, and for heat removal by operable containment equipment. Restoration of emergency cooling should be credited after an appropriate time following the accident.

50(h) Elevated Temperature

This regulatory guide should specify that a mechanistic calculation of the containment atmosphere thermal history be made with appropriate treatment of stratification including consideration of the following: (1) hydrogen combustion, (2) high pressure melt ejection, (3) LOCAs, and (4) molten corium-concrete interaction. A detailed heat transfer analysis should be required to ensure that seals, penetrations, equipment, and other items of safety significance are not damaged. For containments with steel liners, thermal stresses induced by stratification should be considered.

TABLE I

SUMMARY OF PROPOSED CHANGES TO REGULATIONS TO INCORPORATE
SEVERE ACCIDENT CONSIDFRATIONS
INTO CONTAINMENT DESIGN REQUIREMENTS

	Regulation	Subpart	Description of Change	Ancillary Requirements
1	10CFR50 Appendix A General Design Criteria	GDC 16 Containment Design	No change required	Regulatory Guide changed to specify maximum acceptable (e.g., 1%/day) leakage rate for range of severe accident challenges
2	10CFR50 Appendix A General Design Criteria	GDC 38 Containment heat removal	Requirement for heat removal capability consistent with requirements of GDC 50(f) and 50(g)	None
3	10CFR50 Appendix A General Design Criteria	GDC 39 Inspection of containment heat removal system	No change required	None
4	10CFR50 Appendix A General Design Criteria	GDC 40 Testing of containment heat removal system	No change required	None
5	10CFR50 Appendix A General Design Criteria	GDC 41 Containment atmosphere cleanup	Requirement for cleanup system to function consistent with requirements of GDC 50(f)	New RG may be needed
6	10CFR50 Appendix A General Design Criteria	GDC 42 Inspection of containment atmosphere cleanup system	No change required	None

	Regulation	Subpart	Description of Change	Ancillary Requirements
7	10CFR50 Appendix A General Design Criteria	GDC 43 Testing of containment atmospheric cleanup system	No change required	None
8	10CFR50 Appendix A General Design Criteria	GDC 50 Containment design basis	Existing GDC 50 will be expanded by new 50(a)-50(h), described below	As shown below
9	10CFR50 Appendix A General Design Criteria	GDC 50(a) Containment design basis-LOCA	Specifies requirement that containment accommodate LOCA or steam line failure without excessive leakage	New RG needed
10	10CFR50 Appendix A General Design Criteria	GDC 50(b) Containment design basis-fuel-coolant interaction	Specifies requirement that containment accommodate FCI without excessive leakage	New RG needed
11	10CFR50 Appendix A General Design Criteria	GDC 50(c) Containment design basis-hydrogen combustion and detonation	Specifies requirement that containment accommodate hydrogen combustion or detonation without excessive leakage	New RG needed
12	10CFR50 Appendix A General Design Criteria	GDC 50(d) Containment design basis-melt attack	Specifies requirement that containment accommodate direct attack of molten corium without excessive leakage	New RG needed

	Regulation	Subpart	Description of Change	Ancillary Requirements
13	10CFR50 Appendix A General Design Criteria	GDC 50(e) Containment design basis-high pressure melt ejection	Specifies requirement that containment accommodate high pressure melt ejection without excessive leakage	New RG needed
14	10CFR50 Appendix A General Design Criteria	GDC 50(f) Containment design basis-corium-concrete interaction	Specifies requirement that containment accommodate corium-concrete interaction without excessive leakage or contamination of groundwater	New RG needed
15	10CFR50 Appendix A General Design Criteria	GDC 50(g) Containment design basis-pressurization by steam	Specifies requirement that containment accommodate pressurization by steam from decay heat without excessive leakage	New RG needed
16	10CFR50 Appendix A General Design Criteria	GDC 50(h) Containment design basis-elevated temperatures	Specifies requirement that containment accommodate elevated temperatures without excessive leakage or damage to key equipment	New RG needed
17	10CFR50 Appendix A General Design Criteria	GDC 51 Fracture prevention of containment pressure boundary	No change required	None
18	10CFR50 Appendix A General Design Criteria	GDC 52 Capability for containment leakage rate testing	No change required	None

	Regulation	Subpart	Description of Change	Ancillary Requirements
19	10CFR50 Appendix A General Design Criteria	GDC 53 Provisions for containment testing and inspection	No change required	None
20	10CFR50 Appendix A General Design Criteria	GDC 54 Piping systems penetrating containment	Simplify and make consistent with new GDC 58	New RG may be needed
21	10CFR50 Appendix A General Design Criteria	GDC 55 Reactor coolant pressure boundary penetrating containment	Simplify and make consistent with new GDC 58	New RG may be needed
22	10CFR50 Appendix A General Design Criteria	GDC 56 Primary containment isolation	Simplify and make consistent with new GDC 58	New RG may be needed
23	10CFR50 Appendix A General Design Criteria	GDC 57 Closed system isolation valves	Simplify and make consistent with new GDC 58	New RG may be needed
24	10CFR50 Appendix A General Design Criteria	New GDC 58 On-line monitoring of containment isolation status	Reduces likelihood of inadvertent bypass	New RG needed
25	10CFR50 Appendix A General Design Criteria	New GDC 59 Protection of nuclear components against external threats	Alternatively, revise GDC 2 to include structure challenges from aircraft crashes, etc.	New RG needed

	Regulation	Subpart	Description of Change	Ancillary Requirements
26	10CFR50 Appendix A General Design Criteria	New GDC 59-A Assurance of containment integrity during shutdown	Ensure containment designs to permit emergency closure during shutdown operations	New RG needed
27	10CFR100 Reactor Site Criteria	NA	Siting criteria will be uncoupled from containment design criteria in a separate staff program	New criteria/RG as appropriate
28	10CFR50 Appendix J Containment Leakage Testing	NA	Allowable leak rate as established in GDC 16; credit on-line monitoring per new GDC 58	New RG needed
29	10CFR50.44 Combustible Gas Control	NA	Superseded by GDC 50(c) in part	Delete
30	10 CFR 50.34(f) TMI-Related Requirements	NA	Superseded by new GDC 50	Delete

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

July 18, 1991

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

Dear Chairman Selin:

SUBJECT: PROPOSED NUMARC/EPRI FIRE VULNERABILITY EVALUATION METHODOLOGY FOR USE IN THE IPEEE

During the 375th meeting of the Advisory Committee on Reactor Safeguards, July 11-13, 1991, we reviewed the Fire Vulnerability Evaluation (FIVE) Methodology developed by the Nuclear Management and Resource Council (NUMARC) and the Electric Power Research Institute (EPRI) for possible application by licensees in performing their individual plant examination of external events (IPEEE) and the NRC staff evaluation of this methodology. This matter was discussed during a meeting of our Subcommittee on Extreme External Phenomena on July 10, 1991. During this review, we had the benefit of discussions with representatives of NUMARC/EPRI and the NRC staff. We also had the benefit of the documents referenced.

The FIVE methodology has been developed and proposed as an alternative to probabilistic risk assessment (PRA) for identifying potential severe accident vulnerabilities that could result from internal fires at nuclear power plants. In its draft evaluation report, the NRC staff has reviewed this methodology and has identified a number of clarifications and enhancements that they believe would improve the methodology. One of these clarifications, which we believe to be of particular importance, deals with the effect of fire suppressants on safety equipment. This same consideration applies to the alternative PRA methods of fire evaluation. A further improvement, to provide guidelines for compartment interaction analysis, has been agreed to by the proponents.

The NRC staff has concluded that the FIVE methodology, if modified to incorporate these clarifications and enhancements, would be adequate for use in the IPEEE. We agree. This agreement is based, in large part, on our belief that the effectiveness of a search for vulnerabilities will depend as much on the competence and dedication of those making the search as on the particular choice of methodology.

Sincerely,

David A. Ward

David A. Ward
Chairman

References:

1. Memorandum dated May 8, 1991 from W. Minners, Office of Nuclear Regulatory Research, NRC, for R. Fraley, Advisory Committee on Reactor Safeguards, Subject: ACRS Review of NUMARC/EPRI Fire Vulnerability Evaluation (FIVE) Methodology for Use in the IPEEE, with attachments, as follows:

   (a) Draft NRC Staff Evaluation Report on Revised NUMARC/EPRI Fire Vulnerability Evaluation (FIVE) Methodology (undated)
   (b) Letter dated November 14, 1990 from W. Kasin, NUMARC, to W. Minners, NRC, transmitting the following:
      (i) Fire Vulnerability Evaluation Methodology (FIVE) - Plant Screening Guide, Prepared for EPRI by Professional Loss Control, November 2, 1990
      (ii) Fire Events Database for U.S. Nuclear Power Plants, Draft Final Report prepared by SAIC for EPRI, November 26, 1990 (Proprietary)
      (iii) Letter dated November 20, 1990 from J. P. Sursock (EPRI) to D. Modeen, NUMARC, Subject: Comparison Between FIVE Fire Hazard Analysis Methodology and Experimental Data

2. Letter dated May 7, 1991 from R. Ng (NUMARC) to T. King, Office of Nuclear Regulatory Research, NRC, Subject: Response to Draft NRC Evaluation Report
3. Draft NRC Staff Evaluation Report on Revised NUMARC/EPRI Fire Vulnerability Evaluation (FIVE) Methodology (Latest Version Provided to ACRS on July 10, 1991) (Predecisional)

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

December 18, 1991

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

Dear Chairman Selin:

SUBJECT: SECY-91-262, "RESOLUTION OF SELECTED TECHNICAL AND SEVERE ACCIDENT ISSUES FOR EVOLUTIONARY LIGHT WATER REACTOR (LWR) DESIGNS"

During the 380th meeting of the Advisory Committee on Reactor Safeguards, December 12-14, 1991, we considered SECY-91-262, "Resolution of Selected Technical and Severe Accident Issues for Evolutionary Light Water Reactor (LWR) Designs" dated August 16, 1991. Our Subcommittee on Safety Philosophy, Technology, and Criteria discussed this matter on December 5, 1991. We had the benefit of presentations by members of the NRC staff during these meetings and the documents referenced.

SECY-91-262 was prepared by the staff in response to a Staff Requirements Memorandum (SRM) of May 22, 1991, which "requested the staff to provide the advantages and disadvantages of proceeding with generic rulemaking on these issues." The issues in question were not precisely defined in the SRM nor in SECY-91-262, but include fifteen instances, as discussed in SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," dated January 12, 1990, in which the staff proposes to depart from current regulations.

SECY-91-262 cites four advantages for the proactive approach (i.e., generic rulemaking) as summarized below:

(1) Reduced likelihood for litigation in the design certification proceedings by codifying the Commission's policy decisions into enforceable standards.

(2) Better opportunity for the public to participate early in the development of standards.

(3) Facilitation of design certification applications by early clarification and codifying of the Commission's requirements.

(4) Increased confidence of designer-applicants that their submittals can be approved.

The paper also cites four disadvantages, as summarized below:

(1) Generic rulemaking would throw the current schedule for certification of evolutionary designs into disarray.

(2) The diversity of designs will make it difficult to write generic rules with sufficiently detailed criteria.

(3) Additional staff resources will be necessary if generic rulemaking is to be applied to evolutionary designs.

(4) The interdependence of certain complex issues indicates generic rulemaking will be difficult and protracted.

The cited advantages are compelling and well stated. On the other hand, only the first of the cited disadvantages (concern about the schedule) is meaningful. By its approval of the schedule for certifying the evolutionary designs, the Commission effectively ruled out any course other than design-specific rulemaking. The staff proposes to proceed with this course for the ABWR and ABB CE System 80+ designs and states its intent to continue with generic rulemaking activities where appropriate for other evolutionary and passive design applications. In reality, this approach could apply only to passive designs.

The advantages of a generic rulemaking approach are real and important. The design-specific rulemaking process can be carried out in a sound manner, but generic rulemaking is technically preferable. We urge the Commission not to let the opportunity slip by for using this better approach for design certification of the passive plants. We call your attention to our report of May 17, 1991 in which we proposed means by which a proactive approach to severe accident issues could be taken for the passive plant designs.

Sincerely,

David A. Ward

David A. Ward
Chairman

<u>References</u>:

1. SECY-91-262 dated August 16, 1991, for the Commissioners from James M. Taylor, NRC Executive Director for Operations, Subject: Resolution of Selected Technical and Severe Accident Issues for Evolutionary Light Water Reactor (LWR) Designs

2. Staff Requirements Memorandum dated May 22, 1991, for James M. Taylor, NRC Executive Director for Operations, and William C. Parler, NRC General Counsel, from Samuel J. Chilk, Secretary, Subject: Evolutionary Light Water Reactor Certification Issues and Related Regulatory Requirements
3. SECY-90-016 dated January 12, 1990, for the Commissioners from James M. Taylor, NRC Executive Director for Operations, Subject: Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements
4. Report dated May 17, 1991, from David A. Ward, ACRS Chairman, to Kenneth M. Carr, NRC Chairman, Subject: Proposed Criteria to Accommodate Severe Accidents in Containment Design

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

May 17, 1991

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

Dear Mr. Taylor:

SUBJECT. DOCUMENTATION OF COMPUTER CODES

During the 373rd meeting of the Advisory Committee on Reactor Safeguards, May 8-11, 1991, we discussed documentation requirements for computer codes. We previously commented on this matter in a letter to you dated October 11, 1990. This matter was discussed during a meeting of the joint Thermal Hydraulic Phenomena/Severe Accidents Subcommittee held on March 21, 1991. Our Subcommittee on Thermal Hydraulic Phenomena held a meeting on August 28, 1990, in Idaho Falls, Idaho, to review the documentation associated with the RELAP5/MOD3 code developed by the NRC. During these meetings, we had the benefit of discussions with representatives of the NRC staff and its contractor. We also had the benefit of the documents referenced.

At the close of the March 21, 1991 subcommittee meeting, and again at this full Committee meeting, we were asked to comment on a "Charter for Evaluation of RES Code Documentation" to be used as a guide for documentation reviews. In general, we believe the Charter is adequate. However, we recommend adding reference to NUREG-1230, Section 4.4.3, entitled "Code Documentation to Address Scaling and Code Applicability" so that the reviewers apply the lessons learned about documentation requirements from the TRAC-PF1/MOD1 uncertainty study.

We received a memorandum from Eric S. Beckjord, RES, to David A. Ward, ACRS, dated April 10, 1991, with an enclosure entitled "NRC/RES Software Documentation Guidance." Although this guidance is a beginning, it should be fleshed out by providing more explicit guidance concerning the contents of the "Code Manual" and the "Developmental Assessment" document. For example, the "Code Manual" should contain requirements for time-step and nodalization studies dealing with convergence and accuracy. The "Developmental Assessment" document should contain guidance for application of the codes to full-scale nuclear power plants with reference to the convergence and accuracy studies.

To summarize, we recommend the following:

1. The guidelines for code documentation supplied to us by RES should be fleshed out and cited by reference in all code development work statements. Programs to maintain existing codes should include a task to bring code documentation into compliance with the proposed guidelines.

2. A similar set of guidelines should be developed for use by NRR in its review of industry codes used for safety evaluations.

3. Our proposal to modify the Charter for Evaluation of RES Code Documentation review should be adopted.

We would like to be kept informed of progress on this issue.

Sincerely,

David A. Ward

David A. Ward
Chairman

References:

1. Memorandum dated November 23, 1990, from James M. Taylor, Executive Director for Operations, NRC, to Carlyle Michelson, Chairman, ACRS, Subject: NRC Computer Codes and Their Documentation
2. Memorandum dated April 10, 1991, from Eric S. Beckjord, Office of Nuclear Regulatory Research, to David A. Ward, Chairman, ACRS, Subject: NRC/RES Software Documentation Guidance
3. Charter for Evaluation of RES Code Documentation (undated) - Provided to Joint Thermal Hydraulic Phenomena/Severe Accidents Subcommittee during March 21, 1991 meeting
4. U.S. Nuclear Regulatory Commission, NUREG-1230, Subject: Compendium of ECCS Research for Realistic LOCA Analysis, December 1988

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

January 14, 1991

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

Dear Mr. Taylor:

SUBJECT: PROPOSED RESOLUTION OF GENERIC SAFETY ISSUE 29, "BOLTING DEGRADATION OR FAILURE IN NUCLEAR POWER PLANTS"

During the 369th meeting of the Advisory Committee on Reactor Safeguards, January 10-11, 1991, we reviewed the NRC staff's proposed resolution of Generic Safety Issue 29, "Bolting Degradation or Failure in Nuclear Power Plants." Our Subcommittee on Materials and Metallurgy also reviewed this matter during its meeting on January 9, 1991. During this review, we had the benefit of the documents referenced and of discussions with representatives of the NRC staff.

The proposed resolution deals with implementation of a plant-wide bolting integrity program with emphasis on safety systems. The staff's basis for the proposed resolution is described in NUREG-1339. This program has several parts. Some parts involve NRC actions, but most stem from an industry program that is summarized in Electric Power Research Institute report EPRI NP-5769, Volumes 1 and 2.

We agree with the staff that NUREG-1339 provides a satisfactory basis for the proposed resolution of this Generic Safety Issue. The NRC staff has not yet agreed on the method of implementation for this resolution. We withhold final comment on this issue until it is clear what path the NRC staff chooses to follow.

Sincerely,

David A. Ward

David A. Ward
Chairman

References:

1. Memorandum dated December 4, 1990 from Warren Minners, Office of Nuclear Regulatory Research, to Raymond F. Fraley, Advisory Committee on Reactor Safeguards, Subject: Proposed Resolution of GSI-29, "Bolting Degradation or Failure in Nuclear Power Plants."

2. U.S. Nuclear Regulatory Commission, NUREG-1339, "Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants," June 1990.
3. Electric Power Research Institute, EPRI NP-5769, Volumes 1 and 2, "Degradation and Failure of Bolting in Nuclear Power Plants," April 1988.

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

April 18, 1991

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

Dear Mr. Taylor:

SUBJECT: PROPOSED RESOLUTION OF GENERIC SAFETY ISSUE 130, "ESSENTIAL SERVICE WATER SYSTEM FAILURES AT MULTI-UNIT SITES"

During the 372nd meeting of the Advisory Committee on Reactor Safeguards, April 11-13, 1991, we reviewed the NRC staff's proposed resolution of Generic Safety Issue 130, "Essential Service Water System Failures at Multi-Unit Sites." Our Subcommittee on Auxiliary and Secondary Systems also reviewed this matter during its meeting on March 22, 1991. During this review, we had the benefit of discussions with representatives of the NRC staff and of the documents referenced.

We do not agree with the staff's conclusion that issuance of the proposed generic letter has been justified on a cost-benefit basis. A number of assumptions used in the analysis do not appear to provide a fair and balanced comparison of potential costs and benefits. It appears to us that there would be a wide variation in the conclusions if the analysis were done for each individual plant.

We believe that the emergency service water systems of these seven plants should be analyzed as a part of their Individual Plant Examinations (IPEs). Vulnerabilities should be corrected where necessary. The staff should consider making the analysis it has performed for this proposed resolution available to these licensees for use in performing their IPEs.

In the interim, we believe that the staff can assure itself through its inspection program that the licensees of these plants are applying appropriate risk management to the operation and surveillance of their emergency service water systems.

We will consider the advisability of requiring a separate and independent cooling system for reactor coolant pump seals when we review the proposed resolution of Generic Issue 23, "Reactor Coolant Pump Seal Failures."

Sincerely,

David A. Ward
Chairman

References:

1. Memorandum dated March 6, 1991 from Warren Minners, Office of Nuclear Regulatory Research, to Raymond F. Fraley, Advisory Committee on Reactor Safeguards, Subject: Resolution Package of Generic Issue 130, "Essential Service Water System Failures at Multi-Unit Sites," with enclosures (Predecisional)

2. Memorandum dated March 29, 1991 from Warren Minners, Office of Nuclear Regulatory Research, to Raymond F. Fraley, Advisory Committee on Reactor Safeguards, Subject: Resolution Package of Generic Issue 130, "Essential Service Water System Failures at Multi-Unit Sites," with enclosures (Predecisional)

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

May 17, 1991

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

Dear Mr. Taylor:

SUBJECT: PROPOSED RESOLUTION OF GENERIC SAFETY ISSUE 29, "BOLTING DEGRADATION OR FAILURE IN NUCLEAR POWER PLANTS"

During the 373rd meeting of the Advisory Committee on Reactor Safeguards, May 8-11, 1991, we discussed the proposed method for implementation of the resolution of GSI-29 described in your letter of March 21, 1991. That letter states that RES and NRR have agreed to issue a generic information letter, with NUREG-1339 as an enclosure. The proposed generic letter will suggest, but will not require, specific action or written responses from the licensees. The Committee concurs with this method of implementation.

Sincerely,

David A. Ward

David A. Ward
Chairman

References:

1. Letter dated March 21, 1991, from James M. Taylor, EDO, to David A. Ward, ACRS, Subject: Proposed Resolution of Generic Safety Issue 29, "Bolting Degradation or Failure in Nuclear Power Plants"
2. U.S. Nuclear Regulatory Commission, NUREG-1339, "Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants," June 1990

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

August 13, 1991

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

Dear Mr. Taylor:

SUBJECT: PROPOSED RESOLUTION OF GENERIC ISSUE 130, "ESSENTIAL SERVICE WATER SYSTEM FAILURES AT MULTI-UNIT SITES" AND TASK ACTION PLAN FOR GENERIC ISSUE 153, "LOSS OF ESSENTIAL SERVICE WATER IN LWRs"

During the 376th meeting of the Advisory Committee on Reactor Safeguards, August 8-9, 1991, we discussed your May 9, 1991 response to our report to you dated April 18, 1991, on the proposed resolution of GI-130. During this discussion, we also had the benefit of the documents referenced.

Since licensees will be examining their essential service water systems (ESWS) in detail as an important part of their IPE efforts, we agree with your decision to make the analysis used by the staff in its proposed resolution of GI-130 available to licensees. This information should assist them in carrying out their IPEs. We do not, however, understand your statement that "... using the IPE as our vehicle to resolve this generic issue is not a practical option." It seems to us that, if these licensees do a conscientious job of performing their IPEs and identify and correct vulnerabilities involving their ESWS, resolution of the GI-130 issue can be accomplished on a plant-specific basis within a reasonable time.

We believe that the analysis of GI-130 was extremely conservative with respect to the methodology used to establish 1) the frequency of loss of ESWS and 2) the accident mitigation attributes of the "representative plant" for these plants. This was recognized by your contractor, Brookhaven National Laboratory, on page vi of the Executive Summary of NUREG/CR-5526, where the statement is made that "... the service water-related CDF ... is considered to be essentially upper bound."

The ACRS has historically recommended that PRAs be performed on a best-estimate basis and that conservatism then be added when needed to deal with uncertainty for regulatory purposes. (We most recently discussed this issue in our report of July 19, 1991, to

Chairman Selin on the subject of "The Consistent Use of Probabilistic Risk Assessment.") It is clear to us that this principle was not applied to the staff's proposed resolution of GI-130 and is not generally applied by the staff to the cost benefit analysis used for generic issue resolution.

Further, we note that RES has recently developed a Task Action Plan (TAP) for Generic Issue 153, "Loss of Essential Service Water in LWRs." This work represents an expansion of GI-130 to the remaining 99 operating LWRs. The TAP states that the IPEs for the population of operating plants "... may provide information related to the ESW system" and "... may also result in an ESW risk model for each plant, which may be useful for this task." We fail to see how a meaningful IPE can be performed without a detailed evaluation of a plant's ESWS and the accident sequences that could result from partial or complete loss of ESWS.

We believe that GI-153 is well enough defined that it could be resolved on a plant-specific basis as part of the IPE process, and we recommend that this approach be followed. We believe also that there may be other generic issues at a similar stage of development and suggest that work on their resolution could be deferred until enough IPEs have been received and evaluated to determine if the expenditure of staff resources to deal with them as generic issues is warranted. We would like to be kept informed on this matter.

Sincerely,

David A. Ward

David A. Ward
Chairman

References:
1. Memorandum dated May 9, 1991, from James M. Taylor, Executive Director for Operations, to David A. Ward, Chairman, Advisory Committee on Reactor Safeguards, Subject: Proposed Resolution of Generic Issue 130, "Essential Service Water System Failures at Multi-Unit Sites"

2. Memorandum dated July 8, 1991, from Warren Minners, Office of Nuclear Regulatory Research, to Eric Beckjord, Office of Nuclear Regulatory Research, Subject: Task Action Plan (TAP) for Generic Issue 153, "Loss of Essential Service Water in LWRs"

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

October 17, 1991

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

Dear Mr. Taylor:

SUBJECT: PROPOSED RESOLUTION OF GENERIC ISSUE 113, "DYNAMIC QUALIFICATION AND TESTING OF LARGE BORE HYDRAULIC SNUBBERS"

During the 378th meeting of the Advisory Committee on Reactor Safeguards, October 10-12, 1991, we reviewed the NRC staff's proposed resolution of Generic Issue 113. Our subcommittee on Structural Engineering reviewed this matter at a meeting on October 9, 1991. During this review, we had the benefit of discussions with representatives of the NRC staff and industry. We also had the benefit of the documents referenced.

We believe that the proposed resolution of this Generic Issue is appropriate. We would like to be kept informed of progress by the staff and the industry in implementing the actions proposed to resolve this issue.

Sincerely,

David A. Ward

David A. Ward
Chairman

Reference:
Memorandum dated September 5, 1991, from Warren Minners, Director, Office of Nuclear Regulatory Research, NRC, for Raymond F. Fraley, ACRS, transmitting attached documents, including:

a. Draft NUREG/CR-5416, EGG-2571, "Technical Evaluation of Generic Issue 113: Dynamic Qualification and Testing of Large Bore Hydraulic Snubbers," August 1991 and

b. Regulatory Analysis, "Resolution of Generic Issue 113, Dynamic Qualification and Testing of Large Bore Hydraulic Snubbers."

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

December 20, 1991

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

Dear Chairman Selin:

SUBJECT: RESOLUTION OF GENERIC SAFETY ISSUE B-56, "DIESEL GENERATOR RELIABILITY"

During the 380th meeting of the Advisory Committee on Reactor Safeguards, December 12-14, 1991, we reviewed the NRC staff's proposed amendment to the station blackout (SBO) rule, 10 CFR 50.63, and the corresponding revision of Regulatory Guide 1.9 that addresses resolution of Generic Safety Issue (GSI) B-56, "Diesel Generator Reliability." A meeting of our Subcommittee on AC/DC Power Systems Reliability was also held on November 20, 1991 to discuss this matter. We also had the benefit of the referenced documents.

In 1990, the staff proposed resolution of GSI B-56 by issuance of a generic letter requiring licensees to adopt the strictures of proposed Regulatory Guide 1.9, Revision 3, pertaining to the establishment of a diesel generator reliability program. The Committee reviewed this proposed resolution during its 364th meeting in August 1990, and did not support the staff's position, arguing that to do so was an "unjustified imposition of maintenance requirements on licensees, in contravention of the Commission's decision to defer issuance of a maintenance rule...." The Committee also noted that the industry was monitoring the reliability of emergency diesel generators (EDG) pursuant to the requirements of the SBO rule.

The Commission also rejected the staff's proposed resolution. Instead, it directed the staff to develop a rule using a "results-oriented" approach. The staff has done this.

In our view, the proposed rule amendment is unnecessary to ensure adequate diesel generator reliability. We continue to believe that the commitments of the licensees to monitor and maintain diesel generator reliability as specified in the SBO rule, combined with industry initiatives in this regard, are sufficient. If an EDG fails to start, it is industry practice to take appropriate corrective actions. We were told by the NRC staff that statistics compiled by the nuclear industry indicate that the present overall diesel generator reliability level is about 98 percent.

In the course of our discussions with the staff, we were also told that there does not now appear to be a problem with emergency diesel generator reliability, but that there might be one in the future. When asked if the proposed rule would solve a problem if one developed, the response was unclear. In a situation in which both staff and licensees have limited resources, we are reluctant to add to their burden a rule which is designed to solve a problem that does not now exist by means of a proposed solution whose results are uncertain.

In summary, we believe that additional regulation of emergency diesel generators is not warranted and the rule should not be promulgated.

Additional comments by ACRS Members James C. Carroll, Ivan Catton, and Paul G. Shewmon and by ACRS Members Thomas S. Kress and Harold W. Lewis are presented below.

Sincerely,

David A. Ward

David A. Ward
Chairman

Additional Comments by ACRS Members James C. Carroll, Ivan Catton, and Paul G. Shewmon

We do not agree with our colleagues' recommendation and believe that this proposed rule should be issued for public comment. In our view, it represents an appropriate approach to the closure of the station blackout rule and will formalize more reasonable technical specification surveillance testing requirements for EDGs. We further believe that the use of performance-based regulation provides a highly desirable approach to regulation, given the present maturity of the nuclear power industry.

It appears to us that licensees with good EDG maintenance programs and root cause analysis techniques will have no difficulty in staying below any of the proposed trigger values. We note that the failure of an EDG to start is not in general a random event, but an event due to some specific cause that is usually identified and corrected. Proper corrective action will generally improve the reliability of the EDG relative to the reliability it had prior to the event; i.e., the cause of failure to start should be eliminated or greatly reduced. The approach used to evaluate a plant's EDG test data needs to recognize this fact. The small amount of data that is available also must be considered. We believe that the proposed rule strikes a reasonable balance in dealing with these issues.

Additional Comments by ACRS Members Thomas S. Kress and Harold W. Lewis

We support the recommendations but have additional reasons.

The statistical treatment in the proposed action is badly flawed, and is beyond repair. The fundamental problem is that the staff is trying to do something that is mathematically impossible, to derive meaningful reliability information from small numbers of failures. To exaggerate the point only a bit, it is like trying to learn the underlying reliability of an airplane by counting how often it has crashed.

There are so many problems that it is pointless to list them, but here are a couple.

Recall that the only information on which the staff is relying is the number of failures to start. Take as an example the case of the "problem diesel" threshold of 4 failures in the last 25 starts. (The use of prejudicial terms like "problem diesel," "false alarms," "early warning," etc., only obfuscate the issue.) A diesel with a claimed 0.95 reliability, which is maintaining that reliability, will trigger that signal on the average, after 312 efforts to start. But 10 percent of the population will do so in less than 46 starts, and the top 10 percent in more than 705 starts. That is a factor of 15. What kind of threshold is that?

Further, it will take a diesel rated for 0.975 reliability 2534 starts, again on the average, to press this trigger. Since problem status is just as important for a 0.975 diesel as it is for a 0.95 diesel, what is the justification for waiting eight times as long to find out?

A particularly troublesome feature of the proposed rule is the proposal to regard activation of the "double trigger" as a punishable offense. Since even a diesel that is kept at the promised reliability will press the trigger (it just takes a little longer) the staff proposes to punish licensees who have done no demonstrable wrong. That is improper.

It would be easy to go on, but the conclusion is clear -- the proposed rule would be an embarrassment if issued, and the fundamental statistical problem, small numbers of failures, cannot be overcome.

Note that these comments apply to individual diesels or individual sites. It is entirely appropriate to monitor industry-wide diesel experience, where appropriate statistical analysis can yield generic information of value. Further, the thrust toward performance-based regulation is commendable - it just wasn't done well here. It could have been.

References:

1. Memorandum dated November 26, 1991 from A. W. Serkiz, NRC, transmitting Draft Commission Paper, Draft Federal Register Notice, and Draft Regulatory Guide 1.9, Revision 3 "Selection, Design, Qualification, Testing, and Reliability of Emergency Diesel Generator Units Used As Class IE Onsite Electric Power Systems at Nuclear Power Plants" (Predecisional)
2. Staff Requirements Memorandum dated June 26, 1991 from Samuel J. Chilk, Secretary, to James M. Taylor, NRC Executive Director for Operations, Subject: SECY-90-340 - "Diesel Generator Reliability," Resolution of Generic Safety Issue B-56
3. Memorandum dated September 20, 1991 from T. M. Novak, NRC, transmitting AEOD Special Study Report, "Performance of Emergency Diesel Generators in Restoring Power to Their Associated Safety Buses - A review of Events Occurring at Power," AEOD/S91-01
4. Letter dated August 14, 1990, from Carlyle Michelson, ACRS, to Kenneth M. Carr, Chairman, NRC, Subject: Proposed Resolution of Generic Safety Issue B-56, "Diesel Generator Reliability"

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

March 12, 1991

The Honorable Kenneth M. Carr
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Carr:

SUBJECT: PROPOSED RULE ON SELECTION, TRAINING, AND QUALIFICATION OF NUCLEAR POWER PLANT PERSONNEL

During the 371st meeting of the Advisory Committee on Reactor Safeguards, March 7-9, 1991, we discussed the Proposed Rule on Selection, Training, and Qualification of Nuclear Power Plant Personnel. This matter was also discussed during a meeting of our Human Factors Subcommittee on March 6, 1991. During these meetings we had the benefit of presentations by and discussions with members of the NRC staff. We also had the benefit of the documents referenced.

Section 306 of the Nuclear Waste Policy Act of 1982 states that, "The Nuclear Regulatory Commission is authorized and directed to promulgate regulations, or other appropriate Commission regulatory guidance, for the training and qualifications of civilian nuclear power plant operators, supervisors, technicians and other appropriate operating personnel." The Commission considered rulemaking, but in 1984 decided, as an alternative, to permit NUMARC and INPO to develop industrywide improvements to personnel training. INPO developed a comprehensive program to accredit training programs for plant personnel established by each plant licensee. In 1985 the Commission issued a Policy Statement on Training and Qualification of Nuclear Power Plant Personnel that endorsed the INPO-managed training accreditation program, with a proviso that it would be evaluated for effectiveness over an initial two-year period. After this evaluation, the Commission concluded the INPO-managed program was functioning effectively, and in 1988 issued an amended policy statement endorsing continuation of the industry program with some minor changes. However, the Commission's decision to forego rulemaking and substitute a policy statement endorsing an industry program was challenged. In 1990 the U.S. Court of Appeals for the District of Columbia Circuit ordered the Commission to promulgate specific regulatory requirements for training and qualification of nuclear power plant personnel.

The staff has developed a proposed rulemaking package that includes a statement of considerations and proposed additions to 10 CFR Parts 50 and 52. This proposal expands the scope of the rule beyond what is now covered by the policy statement and the INPO-managed training accreditation program. It includes:

- Quality Assurance personnel.
- Training in accident management.
- A requirement that a licensee develop and use a formal procedure for selection of personnel to be trained.

The package, which will eventually include a Regulatory Analysis and a revision to Regulatory Guide 1.8, "Qualification and Training of Personnel for Nuclear Power Plants," neither of which we have seen, is expected to be issued for public comment during April 1991.

Although the rulemaking package is incomplete and may not reflect the final staff position, we offer the following comments:

(1) We do not agree with the staff's proposal to require licensees to develop formal procedures for the selection of personnel to be trained. Although selection is clearly important, we believe that this function is best left to the industry. This option is not permitted by the Court in the areas of training and qualification.

(2) Training and qualification requirements for fire brigade and security personnel at nuclear power plants are given in Appendix R of 10 CFR Part 50 and in 10 CFR Part 73, respectively. Neither rule includes requirements for selection of personnel. This is consistent with our recommendation for selection of other plant personnel.

(3) We agree with the staff's proposal to include requirements in the rule for training and qualification of personnel who will be performing quality assurance functions and personnel who will be responsible for accident assessment and mitigation.

With consideration of these comments, we have no objection to issuance of the rulemaking package for public comment. We would like an opportunity to review the package, including any revision to Regulatory Guide 1.8, after the comment period.

Sincerely,

David A. Ward

David A. Ward
Chairman

References:

1. Draft SECY paper for The Commissioners from James M. Taylor, Executive Director for Operations, Subject: Proposed Rulemaking for Selection, Training and Qualification of Nuclear Power Plant Personnel (Predecisional).
2. Section 306 of Public Law 97-425, Nuclear Waste Policy Act of 1982, "Nuclear Regulatory Commission Training Authorization."
3. Commission Policy Statement on Training and Qualification of Nuclear Power Plant Personnel, published in the Federal Register, March 20, 1985.
4. Commission Policy Statement on Training and Qualification of Nuclear Power Plant Personnel, as amended, November 18, 1988.

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

December 18, 1991

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

Dear Chairman Selin:

SUBJECT: NRC STAFF APPROACH TO THE REVIEW OF THE ROLE OF PERSONNEL AND ADVANCED CONTROL ROOMS IN FUTURE NUCLEAR POWER PLANTS AS DESCRIBED IN SECY-91-272

During the 380th meeting of the Advisory Committee on Reactor Safeguards, December 12-14, 1991, we discussed with the NRC staff its approach to the review of the role of personnel and advanced control rooms in future nuclear power plants as described in SECY-91-272. The Committee, as well as several ACRS subcommittees, has been meeting with the staff, EPRI, and the advanced light water reactor (ALWR) vendors concerning these issues as a part of the Committee's overall review of ALWRs. We also had the benefit of the document referenced.

In our report of November 14, 1991, discussing SECY-91-273, "Review of Vendors' Test Program to Support the Design Certification of Passive Light Water Reactors," we recommended that the staff develop a set of General Human Factors Criteria for ALWRs as a way of defining what is needed in order to make a design certification safety determination. It appears to us that the staff is currently using the man/machine interface requirements of the EPRI ALWR Requirements Document as a major part of this definition. The EPRI Requirements Document contains many requirements that the utility industry considers desirable for efficient and reliable operation of ALWR plants, but that go beyond what is needed for the staff's safety determination. A line needs to be drawn between these two types of requirements.

The staff told us during our meeting that it has revised its thinking on this matter as a result of the plan to deal with this and other issues through the design acceptance criteria (DAC) process. We expect to meet with the staff during January 1992 to discuss its draft DAC for the ABWR control room and will have additional comments based on our review of that document.

Sincerely,

David A. Ward
Chairman

Reference:
SECY-91-272 dated August 27, 1991, for the Commissioners from James M. Taylor, NRC Executive Director for Operations, Subject: Role of Personnel and Advanced Control Room in Future Nuclear Power Plants

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

November 15, 1991

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

Dear Chairman Selin:

SUBJECT: STEAM GENERATOR TUBE REPAIR LIMITS

During the 379th meeting of the Advisory Committee on Reactor Safeguards, November 7-8, 1991, we discussed the NRC's steam generator tube repair limit. Our Subcommittee on Materials and Metallurgy reviewed this matter during a meeting on November 6, 1991 and had the benefit of discussions with representatives of the NRC staff and an EPRI/industry committee for alternate repair limits for steam generators.

The sudden rupture of steam generator tubes due to a transient such as a steam line break or a seismic event needs to be precluded. To prevent such ruptures, the Technical Specifications of a plant define an inspection plan for steam generator tubing. In the Technical Specifications, the plugging limit is expressed in terms of imperfection depth alone, and not in terms of imperfection area. The limit of 40 percent on depth is appropriate for general thinning of the tube wall, or for long cracks. However, it is a poor indicator of reduction in burst pressure if the imperfections are deep pits or flaws that are little wider than they are deep.

A repair limit based on depth alone was appropriate when general wall thinning was a common mechanism of tube degradation. However, as water chemistry has improved over the last decade, it has been much more common for the flaws that develop to be short cracks that are localized in areas such as a support plate, or the tube sheet. It is difficult to find and gauge these cracks.

Analysis, burst tests, and experience in many European nuclear plants show that a few short cracks do not have a significant effect on the burst pressure of a tube, even if the cracks go all the way through the tube wall. It is only when these cracks line up and effectively form a long flaw that they significantly reduce the burst pressure. The continued use of the 40 percent depth limit as a repair limit results in a large effort by the licensees and a significant exposure to workers, and leads to the repair of many tubes that have a negligible risk of failure. We urge that the staff be encouraged to work with the industry to establish more appropriate and generic repair limits in a timely manner.

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

Sincerely,

David A. Ward

David A. Ward
Chairman

Additional Comments by ACRS Member Harold W. Lewis

The instruments used in the tube inspections depend upon the effect of the tube on the inductance and mutual inductance of magnetic coils at frequencies for which the tube thickness is comparable to the skin depth. Such measurements of gross properties are in principle insensitive to the morphology of the cracks, and are in particular not unique indicators of crack depth. The staff is therefore regulating according to a parameter that cannot be uniquely measured. These are instruments which are ancient in concept, and some research attention to the development of more discriminatory instrumentation could help a great deal. It is a mistake to believe one is measuring something that is beyond the capability of the measuring instrument.

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

September 10, 1991

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

Dear Chairman Selin:

SUBJECT: THE STAFF'S RECOMMENDATIONS ON THE REGULATORY IMPACT SURVEY REPORT

During the 377th meeting of the Advisory Committee on Reactor Safeguards, September 5-7, 1991, we discussed the staff's proposed SECY-91-172, Regulatory Impact Survey Report-Final, dated June 7, 1991. This matter was also discussed during our August 8-10, 1991 meeting, and by our Subcommittee on Regulatory Policies and Practices on September 3, 1991. During these meetings, we had the benefit of discussions with representatives of NUMARC, INPO, and the NRC staff. The following comments are made in response to a Staff Requirements Memorandum dated November 6, 1990, containing the request that "the ACRS should review the survey results and proposed corrective actions and provide comments to the Commission."

We have often commended the staff for its initiative in conducting the Regulatory Impact Survey, which provided a substantial body of information and impressions from which regulatory improvements could be generated. SECY-91-172 is a first step toward planning these improvements. It is viewed by both us and the staff as an interim document. It is in large part aspirational, and generally confined to laying out programs to meet the concerns that emerged from the survey. While some of those programs (like training) are already in place, most are not, so this letter must also be regarded as incomplete. We do regard the subject as important, since appropriate response to the survey can set the tone for the interaction of the NRC and its licensees, improving mutual confidence, and thereby contributing to nuclear safety for many years.

The Regulatory Impact Survey and its impact will constitute a major development in nuclear regulation. We therefore think it important that this unique opportunity for substantial improvement be translated into more effective protection of the public health and safety, and into a more productive use of regulatory resources.

There are a number of areas to which we would like to call special attention as the process continues.

1) The Commission has issued clear instructions to the staff, through the Safety Goal Policy Statement, the Severe Accident Policy Statement, and related documents, that regulatory actions should be studied for their conformance with the Safety Goals, and their priorities determined through analysis. We hope to see more evidence of the impact of these directives, as these programs crystallize.

2) One of the well known problems identified in the Survey is that many regulatory initiatives appear as individual items, judged on their own merit by their own proponents within the staff, with inadequate coordination. The staff offers more management control as a solution to this problem, and we will have to see if that is adequate.

3) A major product of the survey was a related concern about the sheer burden of the cumulative impact of the many requirements imposed on the licensees, each of which may be meritorious. The staff response bypasses this issue by concentrating on organization and scheduling, rather than pruning. The problem of setting priorities among issues, so that the most significant (in terms of safety) receive most attention, and the least important none at all, still needs work.

4) One of the major issues is the ill-defined dividing line, in practice, between informal advice and formal direction to a licensee by the staff. The path of least discomfort for a licensee is all too often to simply comply with suggestions from the staff, whether regional or headquarters. Since no one is infallible, and informal advice is often not documented, this is an undesirable arrangement. The staff response to this problem is to step up management involvement, and to enhance staff training. It remains to be seen if this will be enough. Regulation through suggestion is a major problem.

5) There is little in SECY-91-172 dealing with the complex of coherence questions that still bedevil the agency, and which we have addressed in our series of letters on the subject.

6) The question of the technical competence of the staff is an uncomfortable one, but must be raised in an atmosphere in which staff influence is so great. We would like to

see more recognition in the staff that there may be a problem. We do not assert that there is, only that it is an important subject.

7) We are sorry to see the SALP issue decoupled from the others. The staff position is that the Commission has spoken on SALP, so it is a closed matter. We think it is closely related to the questions we are dealing with here, and should be part of the package, if only at the Commission level. We have not retracted our earlier recommendations on SALP.

Since so much of the content of SECY-91-172 is aspirational, and since there is a normal tendency for past good intentions to be swallowed by current problems, we think it especially important that the Commission establish some sort of feedback mechanism, to keep track of progress on these matters. Perhaps it would be appropriate to commit now to a new survey in a few years. Whatever the mechanism, it would be unfortunate to squander this opportunity for progress, through inattention. We would, of course, be happy to help.

Sincerely,

David A. Ward

David A. Ward
Chairman

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

September 11, 1991

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

Dear Chairman Selin:

SUBJECT: NEED FOR ACRS REVIEW OF RECENT STAFF PAPERS

During the 377th meeting of the Advisory Committee on Reactor Safeguards, September 5-7, 1991, we considered several staff papers that have been recently forwarded to the Commission for appropriate action. In all cases copies of the papers were provided to the ACRS office, but there were no requests for ACRS review and comment. While some of these papers may be limited to matters that are largely procedural and for which the Commission would not expect ACRS advice, it is not clear that all are. We believe we should have had an opportunity to comment on these important matters. The Commission may wish to postpone any action it intends to take on these papers until we have had an opportunity to consider the papers. This is an unusual cluster of SECY papers on important issues, and it may not be possible for us to review all of them on an expedited basis. During our 378th meeting, October 10-12, 1991, we will develop a schedule for our review.

The papers of interest, with some brief comments, are listed below:

- SECY-91-229, "Severe Accident Mitigation Design Alternatives for Certified Standard Designs," dated July 31, 1991, with a request for a Commission "notation vote" by August 16, 1991.

  An initial reading indicates that this SECY paper may involve only procedural matters, but we request more time to consider its import.

- SECY-91-262, "Resolution of Selected Technical and Severe Accident Issues for Evolutionary Light Water Reactor (LWR) Designs," dated August 16, 1991 (copy received at ACRS office on August 27), with a request for a Commission "notation vote" by September 3, 1991.

  This paper covers matters to which we have given considerable attention and on which we commented to the Commission in our report of May 17, 1991, "Proposed Criteria to Accommodate

Severe Accidents in Containment Design," and also discussed during a meeting with the Commission on June 7, 1991. In addition, the Commission asked for staff response to the May 17, 1991 ACRS report in its SRM of June 18, 1991. Neither the ACRS report nor the SRM are mentioned in this SECY paper.

- SECY-91-270, "Interim Guidance on Staff Implementation of the Commission's Safety Goal Policy," dated August 27, 1991, with a request for a "negative consent" approval by the Commission by September 12, 1991.

  The ACRS has played a major role in development of the Commission's Safety Goal Policy over the last several years and has previously commented at some length on the implementation plan.

- SECY-91-272, "Role of Personnel and Advanced Control Rooms in Future Nuclear Power Plants," dated August 27, 1991, and forwarded to the Commission for information.

  While this paper is only for information, it concerns an issue, digital computer-based control rooms, in which we have great interest. We believe it is vital that the advanced technology involved with these control rooms be carefully scrutinized during the design certification reviews, and we have concerns about whether the staff is properly equipped to conduct the necessary reviews.

- SECY-91-273, "Review of Vendors' Test Programs to Support the Design Certification of Passive Light Water Reactors," dated August 27, 1991, with a request for "negative consent" approval by the Commission by September 12, 1991.

  This paper is of fundamental importance to the Commission's design certification program, and an in-depth ACRS review would be highly appropriate.

Sincerely,

David A. Ward

David A. Ward
Chairman

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

October 17, 1991

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

Dear Chairman Selin:

SUBJECT: SCHEDULE FOR ACRS REVIEW OF RECENT STAFF PAPERS

During the 378th meeting of the Advisory Committee on Reactor Safeguards, October 10-12, 1991, we considered the schedule for our review of several recent staff papers. These papers were the subject of our letter to you of September 11, 1991, in which we stated our interest in having an opportunity to review these papers and suggested that the Commission consider postponement of action until we had such an opportunity. We also indicated in that letter that we would develop a schedule for our review during the 378th meeting.

With regard to the papers in question and the schedule on which we plan to report our comments, we find SECY-91-229, "Severe Accident Mitigation Design Alternatives for Certified Standard Designs," dated July 31, 1991, largely procedural and plan no further review. We will report on the following papers during our February 1992 meeting:

- SECY-91-262, "Resolution of Selected Technical and Severe Accident Issues for Evolutionary Light Water Reactor (LWR) Designs," dated August 16, 1991.
- SECY-91-270, "Interim Guidance on Staff Implementation of the Commission's Safety Goal Policy," dated August 27, 1991.
- SECY-91-272, "Role of Personnel and Advanced Control Rooms in Future Nuclear Power Plants," dated August 27, 1991.
- SECY-91-273, "Review of Vendors' Test Programs to Support the Design Certification of Passive Light Water Reactors," dated August 27, 1991.

Sincerely,

David A. Ward

David A. Ward
Chairman

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

October 17, 1991

Mr. Eric S. Beckjord, Director
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Beckjord:

SUBJECT: PROPOSED PAPER ON METRICATION POLICY

We appreciate the opportunity to review the proposed policy statement on metrication, which is to be sent to the Commission with a staff recommendation that it be issued for public comment. We recognize the staff's problem in reconciling the requirements of the Omnibus Trade and Competitiveness Act of 1988, which mandates a certain level of movement toward the metric system, and the requirements of safety, which would normally preclude unnecessary changes.

Of course we reserve the right to comment on the final version of the proposed Policy Statement when you finally send it to the Commission, but we have a few comments on the draft.

Generally, we support the staff approach, which is to avoid monumental upheaval, while encouraging the use of the metric system by licensees and applicants, and preparing the NRC staff through education and through cooperative interaction with the industry.

We also support the one specific step proposed, to issue all new actions and supporting documents in dual units. We think one could in fact go the next step without a safety penalty (and possibly with a safety advantage) by always using the metric system for the primary units, with the translation into English units in parentheses. That will facilitate the ultimate transition to a fully metric system.

We call your attention to the footnote on page 12, which purports to explain how one is to carry significant figures from one system of units to another. It is wrong.

Sincerely,

David A. Ward
Chairman

Reference:
Draft SECY paper for the Commissioners from James M. Taylor, Executive Director for Operations, NRC, Subject: Metrication Policy, transmitted by memorandum dated October 3, 1991, from Eric S. Beckjord, Office of Nuclear Regulatory Research to Raymond F. Fraley, ACRS (Predecisional)

cc:
James M. Taylor, EDO

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

May 17, 1991

The Honorable Kenneth M. Carr
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Carr:

SUBJECT: FINAL RULEMAKING TO IMPLEMENT THE EMERGENCY RESPONSE DATA SYSTEM

During the 373rd meeting of the Advisory Committee on Reactor Safeguards, May 8-11, 1991, we discussed with the NRC staff its proposed final rule that would amend 10 CFR Part 50 to establish requirements for the implementation of the Emergency Response Data System (ERDS). This matter was also considered during our 362nd meeting, June 7-9, 1990. We also had the benefit of comments by a representative of NUMARC and of the documents referenced.

We previously commented on the proposed rule in our report of June 12, 1990. In that report, we did not support the proposed ERDS rule, although we acknowledged that it had some positive aspects.

During this meeting, we discussed Mr. James M. Taylor's July 24, 1990 response to the Committee's report in which he stated that the Commission, in approving SECY-80-433, had established the role and responsibility of the agency in nuclear plant accidents and that these have been articulated in NUREG-0728 and Manual Chapter 0502. He said also that the need for "timely, accurate and reasonably complete information about plant conditions, radiation releases and meteorological conditions at the site," as would be provided by ERDS, is fundamental in carrying out that role and that the ERDS rule would not change the NRC role or its responsibilities.

In addition, Mr. Taylor stated that, based on his personal participation in actual responses to emergencies and exercises, "the risks of acting on inadequate or incorrect information are far greater than those associated with the modest amount of information that ERDS can provide."

We were told by the staff and NUMARC that the voluntary program is not expected to result in industrywide participation. The present level of commitment represents about 55 percent of licensed power reactors, and is not expected to significantly increase without the rule.

As a result of our present review, we recommend that this rule be promulgated. However, we continue to have a concern that ERDS might encourage inappropriate involvement of the NRC in the management of future serious accidents. All operational aspects of accident management must be the responsibility of the licensee unless the Commission determines that formal intervention is necessary to protect the public health and safety.

We recommend that substantial experience be obtained with the operation of ERDS at a few plants before it is implemented industrywide.

We have also observed that ERDS may not be available during loss of power events. This suggests that emergency plan exercises should be carried out periodically without the availability of ERDS so that voice transmission of data can be practiced by participants.

We wish to be kept informed by the staff of the experience with ERDS as it is implemented.

Additional comments by ACRS members William Kerr and J. Ernest Wilkins, Jr. and by Harold W. Lewis are presented below.

Sincerely,

David A. Ward

David A. Ward
Chairman

Additional Comments by ACRS Members William Kerr and J. Ernest Wilkins, Jr.

The Committee's report of June 12, 1990 did not support the proposed ERDS rule. We still endorse that position and the justification therefor. We recognize the staff's support and expressed need for the information that they believe will become available with the implementation of the ERDS. However, our fear of inappropriate staff intervention in a serious and unanticipated severe accident continues to outweigh our evaluation of the benefits that might be provided by ERDS. We therefore cannot endorse the rule.

Additional Comments by ACRS Member Harold W. Lewis

I continue to believe that the arguments made in our June 12, 1990 letter remain valid, and do not support this reversal on the part of the Committee. Even the manual chapter on the division of responsibility between NRC and licensee in the event of a serious accident is ambiguous, opening the door to informal management on the part of both on-site and off-site NRC personnel. A central principle of all emergency management is the need for an unambiguous chain of command, and a clear transfer of responsibility when management authority is transferred. If this matter were clearly and unambiguously treated, I would see more merit in the proposed system. ERDS is, after all, a direct descendant of the Nuclear Data Link, for which funds were long denied by the Congress, and which died for exactly these reasons.

References:
1. Memorandum dated April 19, 1991, from E. S. Beckjord, Office of Nuclear Regulatory Research, NRC, to David A. Ward, ACRS, transmitting draft SECY paper on Emergency Response Data System
2. Memorandum dated July 12, 1990, from James M. Taylor, Executive Director for Operations, to Mr. Charles J. Wylie, ACRS, regarding response to ACRS report dated June 12, 1990, Subject: Proposed Rule to Implement an Emergency Response Data System
3. Memorandum dated April 29, 1991, from P. Boehnert, ACRS, to ACRS Members, transmitting (a) SECY-80-433 dated September 16, 1980, Subject: Reports to Congress - NRC Response Plan; Emergency Communication; and Nuclear Data Link, (b) NUREG-0728, Revision 1, Subject: NRC Incident Response Plan, April 1983, (c) NRC Manual Chapter 0502, Subject: NRC Incident Response Plan, June 11, 1987

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

May 20, 1991

MEMORANDUM FOR: Recipients of ACRS Reports

FROM: R. F. Fraley, Executive Director

SUBJECT: ACRS REPORT ON STAFF EVALUATION AND RECOMMENDATIONS ON MAINTENANCE RULEMAKING

The attached revised ACRS report replaces the version dated April 17, 1991. The third paragraph on page 2 and the first paragraph on page 4 have been changed to reflect Commission action taken on SECY-90-094.

Attachment:
ACRS report Revised May 20, 1991, Subject:
Staff Evaluation and Recommendations on
Maintenance Rulemaking

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

Revised: May 20, 1991

The Honorable Kenneth M. Carr
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Carr:

SUBJECT: STAFF EVALUATION AND RECOMMENDATIONS ON MAINTENANCE RULEMAKING

During the 372nd meeting of the Advisory Committee on Reactor Safeguards, April 11-13, 1991, we discussed with the NRC staff their current evaluation and recommendations on maintenance rulemaking for nuclear power plants. Our Maintenance Practices and Procedures Subcommittee met with the staff on this matter on April 10, 1991. During these meetings, we had the benefit of comments by a representative of NUMARC and also had the benefit of the documents referenced.

Given the industry initiatives and the improving trend in industry maintenance practices, we agree with the staff's recommendation contained in SECY-91-XXX that the Commission not proceed with rulemaking but, instead, issue a final Policy Statement on maintenance. We do, however, have a number of comments and recommendations on the version of the staff's proposed final Policy Statement on maintenance that we reviewed.

BACKGROUND

We have commented previously on a maintenance rule in our reports of September 13, 1988 and April 11, 1989. While we agreed that a good maintenance program is necessary to ensure safe and reliable nuclear power plant operation, we opposed the promulgation of the various proposed rules and their accompanying regulatory guides. We presented arguments to support our view that this proposed rulemaking was likely to be counterproductive to improved nuclear power plant maintenance practices. It appeared to us that these practices were continuing to improve as the result of substantial industry initiatives that had been in progress since INPO was established in 1980. We also believe that the Commission's emphasis on maintenance over the past several years has served to stimulate this progress.

In our April 11, 1989 report, we commented that the scope of the proposed rule and its accompanying regulatory guide was excessively broad and suggested a reevaluation of current regulations to determine where overall regulatory emphasis (not just maintenance but all facets of regulation) should be placed on balance-of-plant systems. We also suggested, based on our discussions with the staff, including the Office of the General Counsel, that improvements could be effected for the few plants with "poor" maintenance programs by enforcement of existing regulations.

In our report of October 12, 1989, we commented on a proposed revision to the Commission's March 23, 1988 Policy Statement on nuclear power plant maintenance. We recommended that this revised Policy Statement not be issued, but that the staff obtain additional public and industry comments and continue to monitor industry improvement efforts in order to determine if a rule or Policy Statement was really needed. By doing so, the staff would gain additional information that would be helpful in defining scope and content for a rule or policy statement. We also expressed concern regarding the staff's proposal that an enforcement policy be adopted wherein escalated civil penalties would be imposed for violations where maintenance was the root cause. We pointed out that this might cause licensees to divert resources from other important safety-related activities with a net negative impact on safety.

The Commission issued this revised Policy Statement on December 12, 1989 but requested the staff to inform the Commission of public comments received relative to the escalated enforcement policy that was included in the Policy Statement. The staff, in SECY-90-094 dated March 15, 1990, provided this information and recommended that the escalated enforcement policy be rescinded. The Commission disapproved this recommendation.

THE STAFF'S CURRENT MAINTENANCE RULEMAKING PACKAGE

At this time, the staff is in the process of preparing a SECY document presenting its recommendations. A final position has not been reached, and our review and comments are based on a draft version, marked up to reflect the staff's responses to prior reviews by the CRGR and senior staff management, and further revisions proposed orally by the staff during our meeting.

The staff addresses the need for a maintenance rule and recommends that no rule be promulgated. Instead, the staff recommended that the Commission should issue a revised Policy Statement that emphasizes the need for licensees to complete the ongoing efforts to develop and continue to maintain effective maintenance programs. The proposed SECY also describes the staff's plan for monitoring industry programs. Further, the staff during our meeting proposed its intention to recommend rescission of the present enforcement policy of escalating civil penalties for violations resulting from poor maintenance practices.

ACRS EVALUATION OF SECY-91-XXX

- We are in agreement with the staff's assessment that the industry has made considerable improvement in the quality of nuclear power plant maintenance over the past several years. This is indicated by the results of maintenance team inspections, reinspections, and improving trends in performance indicators and SALP ratings.

- We are impressed by ongoing industry initiatives and commitments to further improve nuclear power plant maintenance. These include the issuance of INPO 90-008, "Maintenance Programs in the Nuclear Power Industry," which is a compilation of INPO's maintenance performance objectives and criteria. The staff has reviewed INPO 90-008 and concluded that it is an acceptable industry maintenance program document delineating necessary program elements. We agree that this document provides appropriate guidance to a utility manager on how to achieve the objectives required for a good maintenance program.

- The draft Policy Statement, under "Maintenance Definition and Process," provides a compilation of "activities and supporting functions that should be considered in a maintenance program." This compilation comes from the staff's draft performance based regulatory guide and the Commission's current Policy Statement. The listing uses language generally similar to but different from that of INPO 90-008. We recommend that this section of the Policy Statement either be deleted or revised to agree with INPO 90-008 in order to avoid confusion as to the Commission's views.

- The draft Policy Statement, in the last paragraph under "Position," describes those structures, systems, and components (SSCs) that licensees should include in their maintenance programs. We have two concerns with the language of the draft SECY document. First, we believe that the scope envisioned for balance-of-plant SSCs is overly broad. The staff told us that it has prepared revised wording to limit the scope for balance-of-plant SSCs to only those SSCs that could directly result in conditions adverse to safety. This revised wording appears to be acceptable. Our second concern is the absence of explicit language to require the inclusion in the scope of licensee's maintenance programs of those nonsafety-related SSCs that are important to the mitigation of severe accidents. We recommend that the Policy Statement be revised to include these programs.

- The staff told us that it plans to recommend that the maintenance escalation factor, which was made a part of the enforcement policy in the revised Policy Statement published on December 8, 1989, be rescinded. As discussed above, we disagreed with the original establishment of this escalation factor in our report of October 12, 1989. We continue to agree with the staff that the maintenance escalation factor should be rescinded.

- The staff plans to continue to monitor the effectiveness of licensee maintenance programs, as described under "Future Actions" in the draft Policy Statement. This monitoring activity appears to be appropriate for the purpose.

Sincerely,

David A. Ward

David A. Ward
Chairman

References:
1. SECY-91-XXX (Draft), Memorandum for the Commissioners from James M. Taylor, Executive Director for Operations, Subject: Staff Evaluation and Recommendation on Maintenance Rulemaking (Predecisional), transmitted by memorandum dated March 14, 1991 from James H. Sniezek, Nuclear Reactor Regulation, to R. Fraley, ACRS
2. Institute of Nuclear Power Operations, INPO 90-008, Revision 01, "Maintenance Programs in the Nuclear Power Industry," dated March 1990 (Proprietary)
3. SECY-90-094, Memorandum for the Commissioners from James M. Taylor, Executive Director for Operations, Subject: Public Comments Received Concerning the Enforcement Policy Revision Involving Maintenance-Related Root Cause, dated March 15, 1990 (Predecisional)

April 17, 1991

The Honorable Kenneth M. Carr
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Carr:

SUBJECT: DRAFT FINAL RULE ON NUCLEAR POWER PLANT LICENSE RENEWAL

During the 372nd meeting of the Advisory Committee on Reactor Safeguards, April 11-13, 1991, we reviewed the draft of the final rule on nuclear power plant license renewal (10 CFR Part 54). Our Subcommittee on Plant License Renewal discussed this matter during its April 8, 1991 meeting. During our consideration of this matter, we had the benefit of discussions with representatives of the NRC staff, NUMARC, and Northern States Power Company. The latter is the licensee for the Monticello Nuclear Generating Plant, which is a lead plant in the license renewal program. We also had the benefit of the document referenced.

The ACRS reported to you on the proposed license renewal rule in its report of April 11, 1990. Since that time, the proposed rule was published for public comment. The staff received 197 comments. It has assimilated information from these comments and information received in a number of interactions with industry and has prepared a draft final rule. The schedule calls for the final rule to be published by June 28, 1991, and for other parts of the rulemaking package, a regulatory guide and a standard review plan, to be published about one year later.

As stated in our April 11, 1990 report, we concur with the approach being taken by the staff in this rulemaking. However, there are two areas of disagreement between the staff and NUMARC that we would like to bring to your attention. The first might require a modification in the draft final rule. The second is related to implementation of the rule.

The first matter is an issue on which we do not have a recommendation except that it should receive your consideration. The draft final rule requires that each applicant for license renewal develop a "compilation" of its Current Licensing Basis. Although it is not precisely clear what this means, it was agreed that it would, at a minimum, include a list of all licensing commitments agreed to by the applicant over the history of its plant. Industry representatives believe this is unnecessary.

The second issue is how implementation of the rule will be limited in scope to concentrate resources for aging management where needed. The rule would require that each applicant develop a list of Systems, Structures, and Components Important to License Renewal (SSCITLR) and then implement an aging management program appropriate for items on that list. The staff's position is that the original SSCITLR list should include all those items in the plant that play a role in meeting any docketed commitment the licensee has made. This would include the original license; commitments related to new rules as they came into being; and commitments made in response to Safety Evaluation Reports, Information Notices, Bulletins, Generic Letters, and Orders.

The industry representatives told us that such a definition of SSCITLR would result in a list that includes 85 to 90 percent of all equipment in the plant. They believe that application of a special aging program to all of these items would be unnecessary and onerous. The process of reducing the initial SSCITLR list to just those items to be covered by a special aging program is critical. Items important to implement other commitments would not thereby be ignored. They would be maintained through the new license period just as they are now.

We believe that selection of those items to be subjected to a special aging program should be based on technical rather than legal argument. Our understanding is that a program of this nature can be developed with the rule as presently drafted. However, implementation will require careful crafting of the regulatory guide and the standard review plan. We would like the opportunity to review these documents before they are issued.

Sincerely,

David A. Ward

David A. Ward
Chairman

Reference:
Memorandum dated March 6, 1991 from Warren Minners, Office of Nuclear Regulatory Research, to Raymond F. Fraley, ACRS, Subject: Final Rule on Nuclear Power Plant License Renewal, with enclosures (Predecisional)

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

April 23, 1991

The Honorable Kenneth M. Carr
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Carr:

SUBJECT: PROPOSED POLICY ISSUES IDENTIFIED IN SECY-91-078, "CHAPTER 11 OF THE ELECTRIC POWER RESEARCH INSTITUTE'S (EPRI'S) REQUIREMENTS DOCUMENT AND ADDITIONAL EVOLUTIONARY LIGHT WATER REACTOR (LWR) CERTIFICATION ISSUES"

During the 372nd meeting of the Advisory Committee on Reactor Safeguards, April 11-13, 1991, we discussed the two Policy Issues identified in SECY-91-078 related to the certification of the Evolutionary Light Water Reactors. Our Subcommittee on Improved Light Water Reactors also discussed these issues on April 9-10, 1991 in its continuing review of the EPRI Advanced Light Water Reactors (ALWR) Requirements Document. During these meetings, we had the benefit of discussions with representatives of the NRC staff and EPRI. We also had the benefit of the documents referenced.

The staff's position regarding the first Policy Issue is that "an evolutionary ALWR design should include an alternate power source to the non-safety loads unless the design can demonstrate that the design margins in the evolutionary ALWR will result in transients for a loss of non-safety power event that are no more severe than those associated with the turbine-trip-only event in current existing plant designs." The staff's major concern is that the ALWR designs are departures from past practice and may result in an increased frequency of shutdowns that require cooling by natural circulation. Presently licensed plants have electrical systems that provide an alternate power source to non-safety loads on shutdown. However, the staff did not substantiate its concerns with respect to the proposed EPRI design requirements.

EPRI claims that the ALWR is designed to safely accommodate shutdown with natural circulation and that the increased frequency of such events is small with this design. The EPRI requirements for the ALWR electrical system design fully meet General Design Criterion (GDC) 17, "Electric Power Systems," and the staff guidance contained in Regulatory Guide 1.32, Revision 2, "Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants." The ALWR electrical power system design is arranged to supply electric power to the plant's safety loads from the main generator, the plant switchyard, an independent transmission line, a gas turbine generator, and the diesel generators. The design uses a generator circuit breaker between the main generator and the step-up transformer and has an improved full turbine load rejection capability. EPRI claims high reliability of electric power to the unit auxiliary transformers and has provided data to support its claim that the benefits derived from adding an alternate power source to the non-safety loads are small and not cost effective. We concur with the EPRI position.

The staff's position regarding the second Policy Issue is based on a misunderstanding of the text of the EPRI requirements. As a result, the staff proposes an additional requirement that "at least one offsite circuit to each redundant safety division should be supplied directly from one of the offsite power sources with no intervening non-safety buses, in such a manner that the offsite source can power the safety buses upon a failure of any non-safety bus." The staff's concern is that routing offsite power to the safety buses through non-safety buses may subject safety equipment to undesirable disturbances on the non-safety buses. Therefore, the staff's position would require the capability to supply safety buses directly from offsite power. The staff did not substantiate its concern. However, the EPRI requirements for ALWR electrical power system design already provide one alternate circuit to each of the redundant safety divisions directly from offsite power. This meets the staff's position. EPRI agreed to clarify the text to document this requirement. EPRI's position is that the direct circuit from offsite to each of the redundant safety divisions should be the backup power supply and the normal supply should be from the plant's auxiliary electric system. We concur with EPRI's position, but do not believe that this should become a regulatory requirement.

Sincerely,

David A. Ward

David A. Ward
Chairman

1. SECY-91-078, Memorandum dated March 25, 1991 for the Commissioners from James M. Taylor, Executive Director for Operations, Subject: Chapter 11 of the Electric Power Research Institute's (EPRI's) Requirements Document and Additional Evolutionary Light Water Reactor (LWR) Certification Issues (Predecisional)
2. U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Draft Safety Evaluation Report on Chapter 11 of

the Advanced Light Water Reactor Requirements Document for Evolutionary Plant Designs, March 1991

3. Electric Power Research Institute, "Advanced Light Water Reactor Requirements Document, Chapter 11 - Electric Power Systems," Issued April 11, 1989

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

July 18, 1991

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

Dear Mr. Taylor:

SUBJECT: CONCERNS RELATED TO THE GENERAL ELECTRIC ADVANCED BOILING WATER REACTOR DESIGN

During the 375th meeting of the Advisory Committee on Reactor Safeguards, July 11-13, 1991, we discussed the status of the Advanced Boiling Water Reactor (ABWR) design, described in the Standard Safety Analysis Report (SSAR), for which the General Electric Company (GE) has applied for design certification in accordance with 10 CFR Part 50, Appendix O. Our Subcommittee on Advanced Boiling Water Reactors also discussed this matter during its meetings on October 31, 1990, and May 30, 1991, with representatives of GE and the NRC staff. We also had the benefit of the documents referenced.

Our previous letter to you concerning the ABWR design was dated November 24, 1989, and conveyed our comments on Module 1 of the Draft Safety Evaluation Report (DSER). Since this letter, we have been kept apprised of the design and the status of the review while awaiting receipt of additional DSERs. The staff now says that DSER preparation by modules will be discontinued in favor of preparation by SSAR chapters and Standard Review Plan (SRP) sections.

To ensure the completeness of our review, it will be necessary to account for any additions or revisions to each DSER as forwarded by a SECY subsequent to issuance of our respective comment letter. An arrangement acceptable to us is needed to ensure the identification of any additions or revisions, and we should agree on an appropriate time for their review. Our comments will not be complete, however, until we have submitted a report to the Commission concerning the final SER on which we expect to comment by mid-November 1992.

Our activities subsequent to the completion of our November 1989 letter have focused on several design concerns that were discussed with GE and the NRC staff in an effort to ensure an early awareness and understanding. We believe that it is appropriate to document them here for timely consideration and resolution in appropriate DSER sections. We expect to have additional items later. We do not expect separate replies to our concerns provided the staff responds in the appropriate DSER.

1. Control Building Flooding

The proposed ABWR design locates the Reactor Building Cooling Water (RBCW) System at the lowest elevation in the control building with the essential 250-V. DC battery rooms immediately above, and the main control room at the next higher elevation. This arrangement places the main control room below ground grade. Our concern with this arrangement is the potential for control building flooding due to an unisolated break in the open-cycle cooling water piping or components inside the building. The ultimate heat sink (cooling pond) is likely to provide sufficient water to flood the building to near ground grade.

2. Physical Separation Barriers

Internal plant flooding and external events such as fire are of major concern if their effects cannot be confined to a single division of required safe-shutdown equipment. We believe that the key to confinement is the provision of an appropriate separation barrier. However, a classical barrier such as the 3-hour-rated fire barrier may not of itself, be sufficient to ensure divisional separation under the combined effects of pressure, heat, smoke, and flooding which accompany a fire and its mitigation. Also, it would appear from the SRP that the effects of delayed suppression on room temperature, pressure, and barrier leakage need to be considered when determining that safe shutdown can be achieved. We remain unconvinced that divisional separation barriers for the ABWR have been adequately prescribed for the range of events and conditions during which they must provide separation.

Of particular concern is a diesel fuel fire which may be subject to delayed suppression in the ABWR diesel generator rooms which are located inside the reactor building. It is not clear how these rooms will be qualified by design or testing to withstand burning fuel if spread across the floor by a fuel line rupture. Furthermore, it is not apparent how the compartment doors will be qualified for this condition or whether they can confine the fuel to the room. If manual mitigation is required, a fire barrier door must be opened. It is not certain that this can be achieved safely or that the external environmental effects of a prolonged opening of the door have been considered.

3. Environmental Protection for Solid-State Electronics

The ABWR makes extensive use of solid-state electronic components for essential protection, control, and data transmission functions. Such components are known to be susceptible to adverse environmental changes, particularly temperature extremes. We are concerned that a number of these components may be located in plant areas where postulated events such as pipe rupture, fire, internal flooding, or loss of room cooling may create an adverse environment. The response of such components to the environmental change may be unpredictable and lead to unacceptable system interactions or responses. The behavior of solid state electronic components in environments created by off-normal or accident situations needs to be considered before the adequacy of any physical separation and environmental control measures can be evaluated.

4. Review of Chilled-Water Systems

The ABWR makes extensive use of large chilled-water systems to provide essential environmental cooling functions including those for the solid-state electronics. Since there is no SRP for chilled-water systems, the staff uses other guidance such as SRP Section 9.2.2 (Reactor Auxiliary Cooling Water Systems) when performing its safety evaluation. This guidance does not include evaluation of the large refrigeration equipment that is required for chilling the closed-cycle cooling water.

The NRC staff and GE need to evaluate the safety implications of chilled-water systems, including performance under varying accident heat loads, loss-of-offsite-power loading characteristics, and ability to restart and function after a prolonged station blackout. The NRC staff should develop appropriate guidance for such reviews by preparing a suitable SRP now.

5. Use of Leak-Before-Break Methodology Outside of Primary Containment

In our report of March 14, 1989 to then NRC Chairman Zech on "Additional Applications of Leak-Before-Break Technology," we expressed our belief that an avenue for consideration of further extension of the leak-before-break (LBB) concept should exist. This is still our position. We are concerned that the NRC staff is not giving serious consideration to GE proposals to extend the concept to systems outside of the primary containment because the staff feels constrained by General Design Criterion 4 which does not propose review of methodology.

We would like to see a renewed effort by GE and the NRC staff to determine if a real potential for substantial safety and/or economic benefits can be realized in applying properly the LBB concept outside of the primary containment.

6. Use of Integral Low-Pressure Turbine Rotors

The catastrophic failure of a low-pressure (LP) turbine rotor can lead to high-energy missiles that are capable of damaging safety-related equipment. The domestic turbine manufacturers (General Electric and Westinghouse) have been using an LP turbine design for large turbine generators consisting of a relatively small-diameter bored shaft with shrunk-on and keyway locked blade ring disks. The manufacturers are now offering an integral LP turbine rotor machined from a single large-diameter forging. A rotor of this design would operate at much higher stresses than the shaft of a shrunk-on disk rotor.

We were told by the Electric Power Research Institute (EPRI) representatives that a decision has not as yet been made with respect to a requirement in the ALWR Utility Requirements Document for boring the LP turbine rotors. Boring has historically been performed to remove impurity inclusions near the forging centerline. Such inclusions are stress risers and have led in the past to a number of catastrophic turbine and generator rotor failures in fossil-fueled power plants. Modern forging practices minimize such inclusions and present-day nondestructive examination and evaluation techniques provide much greater assurance of the soundness of turbine-generator rotors.

The NRC staff should follow this issue closely since the use of integral LP turbine rotors, particularly if they are not bored, will require the development of an entirely new set of preoperational and periodic operational inspection, evaluation, and acceptance requirements to protect against turbine missiles. (The staff should also consider this issue for LP turbine rotor replacement programs for currently operating plants.)

7. Cavity-Floor Area Beneath Reactor Vessel

The layout of the containment for the proposed ABWR design makes use of a cavity floor area beneath the reactor vessel to deal with core/concrete interaction. This area is based on an EPRI requirement of $0.02m^2$ per MWt. If a larger area is required, major changes to the containment sizing and layout may be needed. Timely development of a Commission position on this issue is important not only to this design but also to the design of all Advanced Light Water Reactor designs.

Sincerely,

David A. Ward
Chairman

References:
1. Letter dated August 17, 1989 from Charles L. Miller, Office of Nuclear Reactor Regulation, NRC, to Patrick W. Marriott, General Electric Company, enclosing Draft Safety Evaluation Report Related to the Final Design Approval and Design Certification of the Advanced Boiling Water Reactor, dated August 1989.
2. Letter dated August 7, 1987 from Thomas E. Murley, Office of Nuclear Reactor Regulation, NRC, to Ricardo Artigas, General Electric Company, enclosing GE Advanced Boiling Water Reactor, Licensing Review Bases, dated August 1987.
3. GE Nuclear Energy, Standard Safety Analysis Report, Advanced Boiling Water Reactor, Chapters 1 through 20.

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

September 10, 1991

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

Dear Chairman Selin:

SUBJECT: INSPECTIONS, TESTS, ANALYSES, AND ACCEPTANCE CRITERIA (ITAAC) FOR DESIGN CERTIFICATIONS

During the 377th meeting of the Advisory Committee on Reactor Safeguards, September 5-7, 1991, we discussed the staff's requests for Commission guidance pertaining to ITAAC, contained in SECY-91-178 and SECY-91-210. We had the benefit of presentations by and discussion with members of the NRC staff and representatives of NUMARC, as well as the documents referenced.

The industry and NRC staff appear to have reached an agreement on the general features of ITAAC. However, there are still open questions on the scope and details of ITAAC and the role of the "validation attributes."

In SECY-91-210, the NRC staff requests Commission guidance on an industry proposal that would allow the staff to issue final design approvals (FDAs) for standardized plants prior to staff approval of the proposed ITAAC. While the regulations require an applicant for a design certification FDA to submit proposed ITAAC, the contents of the FDA itself are not specified in 10 CFR Part 52. The staff has identified three possible policy options, including a proposed approach from NUMARC to resolve this issue. For the Advanced Boiling Water Reactor (ABWR), we were told that much work remains to complete the final ITAAC. However, a proposed ITAAC is expected to be submitted to the staff in December 1991, a year before the scheduled issuance of the FDA. Although the staff recommends Option 2, we believe that Option 3 is preferable. Option 3 would allow the staff to issue the FDAs only for the GE ABWR and the CE System 80+ before completing the ITAAC review and approval and then reevaluate the process for future applications.

The adoption of Option 3 should not affect the staff's safety reviews or result in additional backfit constraints on the staff, since the Commission had previously commented in its February 15, 1991 SRM on the provisions of 10 CFR Part 52 by stating that "ITAAC are to provide reasonable assurance that a plant which references the design is built and will operate in accordance with the design certification, and thus are not to be used to reach a final conclusion on any safety question associated with the design. ITAAC should not be used to impose additional design requirements."

Sincerely,

David A. Ward
Chairman

References:

1. SECY-91-178, Memorandum dated June 12, 1991 For the Commissioners from James M. Taylor, Executive Director for Operations, Subject: Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) for Design Certifications and Combined Licenses (Predecisional)
2. SECY-91-210, Memorandum dated July 16, 1991 for the Commissioners from James M. Taylor, Executive Director for Operations, Subject: Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) Requirements for Design Review and Issuance of a Final Design Approval (FDA) (Predecisional)
3. Staff Requirements Memorandum dated February 15, 1991 from Samuel J. Chilk, Secretary, to James M. Taylor, Executive Director for Operations, Subject: SECY-90-377 - Requirements for Design Certification Under 10 CFR Part 52

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

November 14, 1991

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

Dear Chairman Selin:

SUBJECT: NRC STAFF RECOMMENDATIONS FOR REVIEWING, MONITORING, AND APPROVING VENDORS' TEST PROGRAMS TO SUPPORT THE DESIGN CERTIFICATION OF PASSIVE LIGHT WATER REACTORS AS DESCRIBED IN SECY-91-273

During the 379th meeting of the Advisory Committee on Reactor Safeguards, November 7-8, 1991, we discussed the NRC staff's recommendations for reviewing, monitoring, and approving test programs to support the design certification of passive light water reactors (LWRs) as described in SECY-91-273. The Committee had previously been briefed on the major design features of the passive LWRs by the vendors. An enclosure to SECY-91-273 provides an initial assessment of the planned testing program for the Westinghouse AP-600 passive plant. Our Advanced Boiling Water Reactors and Advanced Pressurized Water Reactors Subcommittees held a joint meeting on November 6, 1991, to discuss this matter. During these meetings, we had the benefit of discussions with representatives of the NRC staff and comments by the Westinghouse Electric Corporation on its planned test program for the AP-600 passive LWR. We also had the benefit of the document referenced.

The staff also discussed two SECY papers that are in preparation; one will describe the need for large-scale, full-pressure, integral systems testing for the Westinghouse AP-600, and the other will provide an initial assessment of the planned testing program for the General Electric Simplified Boiling Water Reactor (SBWR). We plan to comment on these SECY papers when they become available.

Our overall conclusion is that the staff is developing a comprehensive program for reviewing, monitoring, and approving vendors' test programs to support the design certification of passive LWRs. Our specific comments are as follows:

1. The staff's intent to initiate an early formal relationship with the vendors to provide review and oversight of their test programs in advance of receipt of their applications for design certification should be fully implemented. This staff

initiative is of considerable importance if the present schedules for design certification of passive LWRs are to be maintained.

2. The staff's program may go beyond what is needed to support the design certification of passive LWRs. Accordingly, we plan to closely follow implementation of items 4 and 5 of the staff's proposed formal review procedure, which state respectively that, "The NRC may require the vendors to perform additional tests beyond those originally approved, if information from other tests or analyses indicates that previous testing and analyses are not adequate to satisfy the 10 CFR 52.47 requirements," and "The NRC may identify additional confirmatory testing to be done at NRC's expense in the vendor's facilities, beyond the testing required for design certification."

3. Although the SECY paper identifies the staff's concerns, there is little to indicate what would be required to allay these concerns or to provide answers to related questions. Before beginning a test program, the staff should spend additional effort to define not only its concerns, but also to identify what information must be obtained in order to allay those concerns and allow licensing action to proceed. Unless this is done, there is little assurance that the results of the test programs will be useful or used.

4. At the time of our meetings, SECY-91-273 had not been released to the public. This hindered our review since Westinghouse was not aware of the staff's concerns relative to its planned test program for the AP-600 plant. The present policy of delaying the issuance of SECY papers relating to the design certification of advanced reactors until the final Staff Requirements Memorandum is made available should be reconsidered. A change in this policy would facilitate the review process of future SECY papers.

5. Staff representatives informed us that the staff is evaluating the need to construct its own test facilities to model the AP-600 plant. We were told that one of the justifications for the NRC constructing its own facilities is a concern that sharing test facilities with Westinghouse to obtain independent data might represent a "conflict of interest." This matter should be reviewed in light of past examples of successful NRC/industry cooperative efforts in reactor safety testing and the expense and potential schedule impacts.

6. Consideration should be given to testing the thermal hydraulic aspects of ATWS scenarios for the AP-600 plant, including the performance of safety and automatic depressurization system

valves and the passive containment heat removal system under ATWS conditions.

7. Consideration should be given to the capabilities of the containment system relative to molten core spreading and core-concrete interaction, steam explosions, hydrogen detonation, direct containment heating, direct attack of molten core on containment structures, and extremely high level temperatures that could occur in certain accident scenarios. The SECY paper describes, under the heading of Severe Accident Performance Tests, a set of investigations of the above listed phenomena that could provide information about containment loading during severe accidents. Further, the SECY paper contains the statement, "The staff recommends that the testing and evaluations detailed above be performed." However, staff representatives told us that this statement was not correct and that the staff does not intend to recommend these tests.

8. The SECY paper being prepared for the SBWR testing program should include consideration of the performance requirements for the primary containment isolation valves associated with the Reactor Water Cleanup/Shutdown Cooling System. These valves should be selected and tested on the basis of their critical need to interrupt large pipe-break flows in a highly reliable manner. If isolation is not achieved, it is necessary to show that the passive core cooling water supplies inside of containment do not drain through a break outside of containment.

9. We are concerned about the issue of human factors in the review of advanced LWR instrumentation and control systems. The staff should begin developing "General Human Factors Criteria," analogous to the "General Design Criteria," contained in Appendix A of 10 CFR Part 50, as a means to prescribe NRC requirements in this area. Some rules are needed for this important area that are understood by both the staff and the advanced LWR vendors.

10. The staff believes that a full-height, high-pressure integral facility simulating the AP-600 plant is needed for confirmatory research and for validation of its computer codes. The staff is concerned about interactions between different aspects of the various passive safety systems as well as operator actions to recover from a plant upset. The staff was not prepared to defend its view. At this time, we are not convinced that such a facility is needed. We will comment further when the staff completes the development of its basis for such a facility.

We wish to be kept informed as the staff implements the program described in SECY-91-273, and plan to review the related SECY papers that the staff has in preparation when they become available.

Sincerely,

David A. Ward
Chairman

Reference:

SECY-91-273, Memorandum dated August 27, 1991 for the Commissioners from James M. Taylor, Executive Director for Operations, Subject: Review of Vendors' Test Program to Support the Design Certification of Passive Light Water Reactors (Predecisional)

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

July 19, 1991

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

Dear Chairman Selin:

SUBJECT: THE CONSISTENT USE OF PROBABILISTIC RISK ASSESSMENT

During the 375th meeting of the Advisory Committee on Reactor Safeguards, July 11-13, 1991, and in earlier meetings, we discussed the unevenness and inconsistency in the use of probabilistic risk assessment (PRA) in NRC. PRA can be a valuable tool for judging the quality of regulation, and for helping to ensure the optimal use of regulatory and industry resources, so we would have liked to see a deeper and more deliberate integration of the methodology into the NRC activities. Our recommendations to this end are directed at problems that took time to develop, and are likely to take a long time to solve.

PRA is not a simple subject, so ther e wide variations in the sophistication with which it is us the various elements of NRC. There are only a few staff members expert in some of the unfamiliar disciplines -- especially statistics -- that go into a PRA, so it is not surprising that there are inconsistencies in the application of the methodology to regulatory problems.

To illustrate the pr lems, let us just list a few of the fundamental aspects of the use of PRA, in which different elements of the staff seem to go their own ways. These are just illustrations, but each can lead to an erroneous regulatory decision.

1. The proper use of significant figures is in principle a trivial matter, but it does provide a measure of a person's understanding of the limitations of an analysis. Yet we often hear from members of the staff who quote core-damage probabilities to three significant figures, and who appear to believe that the numbers are meaningful. It is a rare PRA in which even the first significant figure should be regarded as sufficiently accurate to play an important role in a regulatory decision, but there is something mesmerizing about numbers, which imbues them with misleading verisimilitude.

They deserve respect, but not so much, and it is wrong to err in either direction.

2. Closely related is uncertainty. There is no way to know how seriously to take the results of a PRA without some estimate of the uncertainty, yet we often hear thoroughly unsatisfactory answers (some perhaps invented on the spot) when we ask about uncertainty. One of the advantages of PRA is that it provides a mechanism for estimating uncertainty, uncertainty which is equally present, but not quantified, in deterministic analyses.

3. Conservatism. A PRA should be done realistically. The proper time to add an appropriate measure of conservatism is when its results are used in the regulatory process. If the PRA itself is done with conservative assumptions (more the rule than the exception at NRC), and is then used in a conservative regulatory decision-making process, self-deception can result, or resources can be squandered.

   The inconsistent use of conservatism was illustrated by a pair of briefings at our April 1991 meeting, which included updates on proposed rules on license renewal and on maintenance. In the former case, we were told that a licensee could use PRA to add an item for later review, but never to remove one -- a one-way sieve. In the latter case we were told that PRA could be used to justify either enhancement or relaxation of maintenance requirements. Foolish consistency may be a hobgoblin, as Emerson said, but there is nothing foolish in seeking consistency in regulation.

4. The bottom line. It has been widely recognized since WASH-1400 that the bottom-line probabilities (of either core melt or immediate or delayed fatalities) are among the weakest results of a PRA, subject to the greatest uncertainties. (That doesn't mean they are useless, only that they should be used with caution and sophistication.) Yet we find staff members unaware of these subtleties, often dealing with small problems, justifying their actions in terms of the bottom-line probabilities. This is only in part due to the Backfit Rule, which almost requires such behavior; it is also inexperience and lack of sensitivity to the limitations of the methodology.

   A number of staff actions and proposals use bottom-line results of a PRA as thresholds for decision making, often with the standard litany about the uncertainty in the reliability of these results. In fact, the quantified uncertainty in the bottom-line results of a PRA is just as important a number as the probability itself. It would be straightforward to employ a decision-making algorithm that <u>prescribes</u> a confidence level for the decision, and uses both the bottom-line probability and the uncertainty to achieve this. A further improvement would be to incorporate the consequences of erroneous decisions, what statisticians would call the loss function, into the decision-making process. The Commission has come close to this approach in its recent instructions to the staff on the diesel generator reliability question.

These are just a few examples of problems with the use of PRA in NRC, all common enough to be disturbing, and increasing in frequency as the use of PRA increases. It has been more than fifteen years since the publication of WASH-1400, a pioneering study which, despite known shortcomings, established the NRC at the forefront of quantitative risk assessment. One could have hoped that by now a coherent policy on the appropriate use of PRA within the agency, on both large and small problems, could have evolved.

We recommend that:

A. A mechanism be found (perhaps a retreat) through which the few PRA and statistical experts now scattered throughout the agency (and generally ignored) can be brought together with the appropriate senior managers and outside experts, to work toward a consistent position on the use of PRA at NRC. It could be worth the time expended. (Among other long-term benefits, such an interaction would add an element of horizontal structure to the NRC's predominantly vertical organization.)

B. The Commission then find a way to give credence and force to that position.

C. The Commission emphasize recruitment of larger numbers of professionals expert in PRA and statistics.

D. The Commission consider some kind of mandate that any letter, order, issue resolution, etc., that contains or depends on a statistical analysis or PRA, be reviewed by one of the expert PRA or statistical groups.

We do not pretend that this is an easy problem. The solution involves not only a cultural shift, so that those few experts already at NRC have some impact, but also substantial enhancement of the staff capabilities. That will require incentives that only the Commission can supply. It is interesting that the Commission's Severe Accident Policy Statement, dated August 1985, stated that "within 18 months of the publication of this severe accident statement, the staff will issue guidance on the form, purpose and role that PRAs are to play in severe accident analysis and decision making for both existing and future plant designs...."

Additional comments by ACRS Members Harold W. Lewis and J. Ernest Wilkins are presented below.

Sincerely,

David A. Ward
Chairman

<u>Additional Comments by ACRS Members Harold W. Lewis and J. Ernest Wilkins</u>

We thoroughly endorse this letter, and regret only that the Committee chose to ignore the parallels between the PRA problems and those in a number of other newer technologies significant to nuclear safety. Recommendation C should have included mention of some of these -- electronics and computers, for example -- which are of increasing importance. Weaknesses in those areas also need correction. Computerized protection and control systems, in particular, require the kind of sophisticated review that NRC is in no position to provide.

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

August 13, 1991

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

Dear Mr. Taylor:

SUBJECT: EVALUATION OF RISKS DURING LOW POWER AND SHUTDOWN OPERATIONS OF NUCLEAR POWER PLANTS

During the 376th meeting of the Advisory Committee on Reactor Safeguards, August 8-9, 1991, we continued our discussion of the NRC staff program to address the risks posed by nuclear power plants during low power and shutdown operations. We had previously received a status report on this program from the staff during our 374th meeting, June 6-7, 1991. During the same meeting, we also heard a presentation from NUMARC concerning industry efforts to address this issue. Our Joint Subcommittee on Plant Operations and Probabilistic Risk Assessment had met with representatives of the staff and NUMARC on June 5-6, 1991, concerning this matter. We also had the benefit of the documents referenced.

We share the staff's concern that this issue needs to be addressed in a thorough and systematic manner and are favorably impressed with the approach being taken. We are encouraged that the industry is also actively pursuing this issue.

There are three aspects of the staff's shutdown risk study that we believe merit comment:

1. The staff was unable to provide us with the information concerning the design of containment equipment hatches that we had requested during our review of NRC Generic Letter 88-17 on loss of decay heat removal. We had asked how many plants have hatches that are pressure-seating and could be easily closed if the containment were in danger of being pressurized, as opposed to plants having pressure-opening hatch designs that require essentially full bolting to accomplish sealing under pressure. This appears to us to be an important question that could be answered by referring to available information. A related issue concerns the ability of the licensees to effect closure of their equipment hatches when AC power is not available. The March 1990 loss-of-power event that occurred at Vogtle, Unit 1, demonstrated the importance of this consideration. The NRC staff has stated

that these matters will be addressed as part of the shutdown risk study.

2. One component of the shutdown risk study is the development of two PRAs designed to quantify risks posed by low power and shutdown operations. The two plants, Surry and Grand Gulf, chosen for these studies are among those previously modeled as part of the NUREG-1150 studies. We pointed out to the staff that neither of these plants is a good surrogate for the U.S. population of operating reactors. Surry is one of the few PWRs that has isolation valves in its reactor coolant system which permits the licensee to minimize operation at "mid-loop" conditions. Grand Gulf represents the BWR/6 product line; as such, it is representative of only a small fraction of the total population of operating BWRs.

   The staff acknowledged this point, but argued that the review of these plants in the NUREG-1150 effort aids in evaluation of shutdown risk. The willingness of the owner/operators to participate in this study was also a consideration. The degree to which these plants can be considered representative of their surrogate populations will need to be established if the shutdown PRA studies are to be relied on in making regulatory decisions concerning the resolution of this issue.

3. Another concern deals with the NRC staff's modeling approach for the PRA studies. The staff has a two-pronged effort under way. For the short term, a coarse "screening analysis" using "conservative" assumptions will be performed on a schedule that supports the staff's commitment to provide recommendations by the end of the year on measures to minimize shutdown risk. For the long term, a more complete PRA study will be conducted. The long-term effort will not be complete at least until some time during 1992-93.

   The staff's discussion of the conservatism being used in these screening analyses raised concerns with us as to the usefulness of this work. For example, we were told that modeling of human error would be dealt with by assuming that, in most cases, the operator makes the wrong decision in taking action during sequences that could lead to core damage. Since these studies will presumably play some role in the recommendations that the staff will present later this year concerning amelioration of shutdown risk, we caution that PRAs performed in this manner can lead to badly flawed regulatory decisions.

   Our views on the use of PRA in the regulatory process are further discussed in our report of July 19, 1991, to Chairman Selin. We recommend that the staff carefully consider the comments presented in that report.

We wish to be kept informed regarding the resolution of the above matters, and we will continue to monitor the progress of the staff and industry programs.

Sincerely,

David A. Ward
Chairman

References:

1. Memorandum dated October 22, 1990, from J. Taylor, Executive Director for Operations, NRC, for the Commissioners, Subject: Staff Plan for Evaluating Safety Risks During Shutdown and Low Power Operation.
2. Memorandum dated September 5, 1990, from J. Taylor, Executive Director for Operations, NRC, for the Commissioners, Subject: Shutdown Risks in Evolutionary and Advanced Reactors.

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

December 14, 1991

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

Dear Chairman Selin:

SUBJECT: THE CONSISTENT USE OF PROBABILISTIC RISK ASSESSMENT

In a letter to us dated October 1, 1991, the Executive Director for Operations (EDO) disagreed with most of the observations in our letter to you of July 19, 1991, on consistency in the use of PRA. We assume that he does not speak for the Commission, but do feel a need to clarify our meaning. We will key this letter to the seven bullets in his letter, but note that we do not agree with his final conclusion that "it does not appear that major problems now exist in the use of probabilistic risk assessment by the staff." The four recommendations at the end of our letter were, of course, addressed to you, so he did not mention them.

- He seems to have misunderstood our concerns about the uneven level of sophistication, and thinks we were addressing the level of complexity. Of course we were not suggesting that one do a complete NUREG-1150 study on each minor issue. Our concern was with instances of low quality, not the page count. We do not agree that the current pattern is "entirely appropriate."

- He makes an unclear distinction between point estimates and best estimates, and states that when "no data are available ... only conservative estimates are possible." That is not correct. There are no conditions under which conservative estimates are appropriate to an analysis designed to reveal actual risk. Conservative estimates are appropriate only for bounding analyses, but this has nothing to do with the availability of the data. This has been a problem for years, and apparently still is.

- Here he deals with uncertainty analysis much as in the first bullet, again using the word "appropriate." See our comments above.

- He says that the staff is "well aware of the uncertainty and unreliability of PRA," but uncertainty and unreliability are two entirely different concepts. We never used the word unreliability. Further, he states that it is not "practical at this time" to move toward formal decision-making algorithms in the cases in which it is possible.

- We commend his efforts to improve the PRA capabilities of the staff. We hope it bears fruit.

- In response to our observation about the need for staff enhancement in these skills, he says that personnel with the relevant backgrounds are at a premium, but that he is trying. But he also says that "staff resources must be carefully prioritized to optimize their influence." That is subject to many interpretations, ranging from a platitude to a statement that he doesn't believe this subject is important. We have seen recent NRC recruiting ads with a list of disciplines needed, and these are not among them.

- He says that he is working to recruit people with expertise in digital instrumentation and control systems. However, one of his senior managers told us last month that the staff had adequate expertise and needed no more.

We ask only that you note these observations, and pass them on to the EDO.

We do note that a middle-level management group is currently being organized to review the staff's PRA activities. We recommended a much more ambitious approach to you, but even in this one we urge you to make sure that it includes some of the few statisticians on the staff.

Sincerely,

David A. Ward

David A. Ward
Chairman

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

December 18, 1991

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

Dear Chairman Selin:

SUBJECT: SECY-91-270, "INTERIM GUIDANCE ON STAFF IMPLEMENTATION OF THE COMMISSION'S SAFETY GOAL POLICY"

During the 380th meeting of the Advisory Committee on Reactor Safeguards, December 12-14, 1991, we considered SECY-91-270, "Interim Guidance on Staff Implementation of the Commission's Safety Goal Policy," dated August 27, 1991. Our Subcommittee on Safety Philosophy, Technology, and Criteria discussed this matter on December 5, 1991. During these meetings, we had the benefit of presentations by members of the NRC staff and of the documents referenced.

SECY-91-270 was prepared by the staff in response to a Staff Requirements Memorandum (SRM) of June 15, 1990 that directed the staff to establish a formal mechanism for ensuring that future regulatory initiatives are evaluated for conformity with the Commission's safety goals.

The ACRS has, in the past, provided extensive comments on implementation of the Safety Goal Policy (Reference 5). Many of the Committee's proposals have been endorsed by the Commission and were available to the staff in developing the procedure proposed in SECY-91-270.

The proposed procedure does not fulfill the Commission's wish for a mechanism to ensure that proposed regulatory initiatives will be tested against the safety goal. It does not incorporate the concept of thresholds defining "how safe is safe enough" which is the heart of the policy. The proposed procedure instead uses only some elements of the safety goal in a screening process to provide guidance in a determination of whether a proposal would provide "substantial additional protection" in the context of the Backfit Rule, 10 CFR 50.109. It then uses a cost-benefit analysis to decide whether implementation of a proposal is warranted. A cost-benefit test thus becomes, in effect, the safety goal. We believe that is not the intent of the Safety Goal Policy.

Although we believe regulations should be subject to cost-benefit considerations, we regard the safety goal as essentially doing that. Cost and benefit considerations were a part in the original determination of safety goal guidelines, e.g., the 0.1 percent health effect values. We might find an argument for lower-level application of cost-benefit analysis, as proposed in SECY-91-270, more persuasive if, in fact, real costs and real benefits were being evaluated. However, benefits quantified in the analysis tend to be dominated by a value ascribed to averted health effects, typically $1000 per person-rem. This value is every bit as arbitrary as the 0.1 percent health effect guidelines. In both instances, the values are intended to be broad surrogates for a number of deleterious offsite effects that could result from accidental releases of radioactive material. However, it is better to leave the cost-benefit considerations at the upper level of the safety goal hierarchy, in effect, to enter the swamp of cost-benefit analysis only once rather than time after time with each regulatory action.

As we have said many times before, we believe the Commission has shown outstanding leadership and vision in adopting the Safety Goal Policy. Practical means for implementing the Policy are needed. Delays have already been too long. We regret that our disagreement with the approach proposed by the staff may cause further delay. We are giving consideration to developing an alternative implementation plan within the next few months that will be agreeable to all.

We note that the proposed procedure is only looking forward; it is intended for application to new regulatory initiatives. In an earlier report, we recommended that a plan be developed for review of the existing body of regulations and regulatory activities against the Safety Goal Policy. We recognize this will be a difficult undertaking. A means to focus resources will be most critical. We understand such an effort is under way within the staff, and we look forward to an opportunity to review any proposal when that is appropriate.

Sincerely,

David A. Ward

David A. Ward
Chairman

References:
1. SECY-91-270 dated August 27, 1991, for the Commissioners from James M. Taylor, NRC Executive Director for Operations, Subject: Interim Guidance on Staff Implementation of the Commission's Safety Goal Policy

2. Staff Requirements Memorandum dated June 15, 1990, for James M. Taylor, Executive Director for Operations, from Samuel J. Chilk, Secretary, Subject: SECY-89-102 - Implementation of the Safety Goals
3. SECY-89-102 dated March 30, 1989, for the Commissioners from Victor Stello, NRC, Executive Director for Operations, Subject: Implementation of Safety Goal Policy
4. Memorandum dated November 16, 1990 for Carlyle Michelson, Chairman, ACRS, from James M. Taylor, NRC, Executive Director for Operations, Subject: Update on Staff Activities for Safety Goal Implementation
5. Reports by the Advisory Committee on Reactor Safeguards on Implementation of the Safety Goal Policy:
   (a) Implementation of the Safety Goal Policy dated 9/11/90
   (b) The Relationship of the Quantitative Safety Goal to the Concept of Adequate Protection dated 11/20/89
   (c) Comments on the Safety Goal Policy and Its Relationship to the Concept of Adequate Protection dated 10/11/89
   (d) Further Comments on Implementation of the Safety Goal Policy dated 2/16/89
   (e) Program to Implement the Safety Goal Policy - ACRS Comments dated 4/12/88
   (f) ACRS Comments on an Implementation Plan for the Safety Goal Policy dated 5/13/87
   (g) Application of NRC Safety Goals in Licensing Issues dated 11/10/86

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

May 17, 1991

The Honorable Kenneth M. Carr
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Carr:

SUBJECT: PROPOSED FINAL RULE REVISING 10 CFR PART 55, "OPERATORS' LICENSES" TO INCLUDE FITNESS-FOR-DUTY REQUIREMENTS

During the 373rd meeting of the Advisory Committee on Reactor Safeguards, May 8-11, 1991, we heard presentations from the staff and NUMARC concerning the staff's proposal to promulgate a final rule revising 10 CFR Part 55, "Operators' Licenses," to include fitness-for-duty requirements, and to modify Appendix C of 10 CFR Part 2, "General Statement of Policy and Procedures for NRC Enforcement Actions," to reflect enforcement sanctions. We also had the benefit of the document referenced.

In our report of December 20, 1989, we concurred with the staff's plan to issue this proposed rule for public comment. This proposed final rule includes the staff's evaluation of public comments.

This proposed rule, which the staff prepared in response to a Staff Requirements Memorandum dated March 22, 1989, would amend 10 CFR Part 55 so that the conditions and cutoff levels established pursuant to 10 CFR Part 26, "Fitness for Duty Programs," become applicable to licensed operators as a condition of their licenses. The proposed rule will provide a basis for taking enforcement actions (as described in the proposed modifications to Appendix C of 10 CFR Part 2) against licensed operators who (1) use drugs or alcohol in a manner that would exceed the cutoff levels contained in the fitness-for-duty requirements of 10 CFR Part 26; (2) are determined by a facility medical review officer to be under the influence of any prescription or over-the-counter drug which could adversely affect his or her ability to safely and competently perform licensed duties; or (3) sell, use, or possess illegal drugs.

We question the need for this rule. The fitness-for-duty requirements of 10 CFR Part 26 apply to all nuclear power plant personnel (including licensed operators), and the existing bases under 10 CFR Part 55 are available to the NRC for taking enforcement action against licensed operators for violation of fitness-for-duty requirements. Although there were nineteen Part

26 fitness-for-duty incidents involving licensed operators during 1990, the staff did not present any arguments that promulgation of this rule would have had an effect on this situation.

We are also concerned that promulgation of this rule will undercut industry's ongoing efforts to enhance the professionalism of all nuclear power plant personnel. The proposed rule appears to unnecessarily challenge the trustworthiness of licensed operators.

We recommend that this proposed rule not be issued. We believe that the industry has undertaken a substantial effort to deal with the difficult issue of fitness for duty and should be given the opportunity to demonstrate the effectiveness of its programs.

Sincerely,

David A. Ward

David A. Ward
Chairman

Reference:
Memorandum dated April 11, 1991, from Jack W. Roe, Office of Nuclear Reactor Regulation, NRC, to Raymond F. Fraley, ACRS, Subject: Revision of 10 CFR 55 to Require Compliance with Fitness-For-Duty Programs and Conforming Modification to Commission's Enforcement Policy

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

May 17, 1991

The Honorable Kenneth M. Carr
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Carr:

SUBJECT: PROPOSED FINAL REVISION TO APPENDIX J TO 10 CFR PART 50 AND RELATED FINAL REGULATORY GUIDE

During the 373rd meeting of the Advisory Committee on Reactor Safeguards, May 8-11, 1991, we considered the proposed revision to Appendix J to 10 CFR Part 50, "Leakage Rate Testing of Containments of Light-Water-Cooled Nuclear Power Plants," and a related Regulatory Guide (Task No. MS 021-5), "Containment System Leakage Testing." These proposals were discussed during a joint Regulatory Activities and Containment Systems Subcommittee meeting on May 8, 1991. During these meetings, we had the benefit of discussions with representatives of the NRC staff and of the nuclear industry. We also had the benefit of the document referenced.

We offer the following findings:

- Revision of Appendix J to 10 CFR Part 50 is desirable.

- The staff's proposal to make the revised version of Appendix J less prescriptive and to provide detailed guidance in a regulatory guide is appropriate.

- The implementation of the proposed revision to Appendix J clearly is a backfit.

- The staff has been unable to conclude that the proposed revision will provide a substantial increase in safety.

- The staff believes that the proposed revision will not increase costs to licensees; some licensees believe otherwise.

- There has been continuing constructive dialogue between the staff and industry representatives, chiefly relating to a Licensing Topical Report being prepared by the BWR Owners' Group. There are still some technical issues that would benefit from further dialogue between the staff and industry.

We understand from the staff that this dialogue will take place prior to issuance of this proposed revision.

- The proposed revision does not reflect new insights and knowledge about the role of containment, and containment leakage, in mitigating the consequences of severe accidents.

In view of these findings, we have no objection to the proposed revision to Appendix J to 10 CFR Part 50 or to the accompanying Regulatory Guide.

Sincerely,

David A. Ward
Chairman

Reference:
Note dated April 9, 1991 to S. Duraiswamy, ACRS, from G. Arndt, RES, Subject: 10 CFR 50, Appendix J and Regulatory Guide MS 021-5, with enclosures:

(a) Draft Federal Register Notice -- Statement of Consideration and Final Appendix J Rule
(b) Draft Federal Register Notice - Statement of Availability and Final Regulatory Guide MS 021-5

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

October 17, 1991

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

Dear Mr. Taylor:

SUBJECT: REGULATORY GUIDES BEING DEVELOPED IN SUPPORT OF THE REVISED 10 CFR PART 20

During the 378th meeting of the Advisory Committee on Reactor Safeguards, October 10-12, 1991, we discussed the four referenced draft Regulatory Guides, related to the implementation of the revised 10 CFR Part 20, "Standards for Protection Against Radiation," for which we have the lead responsibility. Our Subcommittee on Occupational and Environmental Protection Systems and a Working Group of the Advisory Committee on Nuclear Waste (ACNW) discussed these guides, together with eight other guides in this area for which the ACNW has the lead responsibility, during a joint meeting on September 23 and 24, 1991. During this review, we had the benefit of discussions with representatives of the NRC staff and NUMARC and of the documents referenced.

This letter summarizes our general comments on these four proposed Regulatory Guides. Detailed discussions of our concerns occurred during the September 24, 1991 session, and are available in the transcript of that meeting. We understand that we will have the opportunity to review these guides after the public comments have been reconciled.

1. Draft Regulatory Guide DG-8004, "Radiation Protection Programs for Nuclear Power Plants"

   This proposed guide collects and organizes material in one place that has been previously published in several other regulatory guides, for example, Regulatory Guides 8.2, 8.8, and 8.10. This proposed guide covers the important features of a radiation protection program for nuclear power plants. On the other hand, existing power plants already have radiation protection programs that presumably meet the intent of the revised 10 CFR Part 20. It is not evident that the advantages of this consolidated approach outweigh the disadvantage of creating an additional set of criteria for judging licensee performance.

2. Draft Regulatory Guide DG-8006, "Control of Access to High and Very High Radiation Areas in Nuclear Plants"

   This proposed guide not only offers guidance on the implementation of Sections 1101, 2102, 1601, and 1602 of the revised 10 CFR Part 20, but also collects other recommendations contained in earlier NRC Bulletins that were issued after mishaps, or near mishaps, in high and very high radiation areas. This proposed guide adequately covers the important features of a program for controlling access to such areas. It also provides (new) guidance on diving operations. As in the preceding paragraph, much of the material in this proposed guide is available elsewhere, for example, technical specifications and the rule itself. Nevertheless, we believe a regulatory guide, generally written as proposed, will be somewhat beneficial.

3. Draft Regulatory Guide 8.N6, "Planned Special Exposures"

   This proposed guide offers guidance on the implementation of Section 1206 of 10 CFR Part 20 (as well as some other related sections) dealing with infrequent, pre-planned radiation exposures in excess of routine regulatory limits, deemed necessary because of some exceptional circumstances. This proposed guide addresses the important features of such planned special exposures. We believe that it will be useful to issue a regulatory guide, generally written as proposed, in this area.

4. Draft Regulatory Guide 8.7, Revision 1, "Instructions for Recording and Reporting Occupational Radiation Exposure Data"

   This proposed revision contains instructions on the recordkeeping and reporting requirements of the revised 10 CFR Part 20, with detailed instructions on filling out NRC Form 4 (Occupational Radiation Exposure History) and NRC Form 5 (Occupational Exposure Record for Current Year). We do not have an opinion on the substance of the proposed guide, but do agree with the staff that a guide is necessary.

We are aware that the Advisory Committee on Nuclear Waste plans to send you a letter on its review of the regulatory guides in support of the revised 10 CFR Part 20. We agree with its conclusion that the scheduled date for implementation of the revised regulation may be unrealistic.

Sincerely,

David A. Ward

David A. Ward
Chairman

References:
1. U.S. Nuclear Regulatory Commission, Draft Regulatory Guide DG-8004, "Radiation Protection Programs for Nuclear Power Plants," September 1991
2. U.S. Nuclear Regulatory Commission, Draft Regulatory Guide DG-8006, "Control of Access to High and Very High Radiation Areas in Nuclear Power Plants," September 1991
3. U.S. Nuclear Regulatory Commission, Draft Regulatory Guide 8.N6, "Planned Special Exposures," August 7, 1991
4. U.S. Nuclear Regulatory Commission, Draft Regulatory Guide 8.7, Revision 1, "Instructions for Recording and Reporting Occupational Radiation Exposure Data," September 17, 1991

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

January 15, 1991

The Honorable J. Danforth Quayle
President of the United States Senate
Washington, D.C. 20510

Dear Mr. President:

In accordance with the requirements of Section 29 of the Atomic Energy Act of 1954, as amended by Section 5 of Public Law 95-209, the Advisory Committee on Reactor Safeguards has reported to the Congress each year on the Safety Research Program of the Nuclear Regulatory Commission. In our December 18, 1986 letter to the Congress, we proposed to provide reports on specific issues rather than one all-inclusive annual report.

During the past year, we have reviewed the NRC Safety Research Program and other closely related matters in the following areas:

- Nuclear Power Plant Containment Performance Improvement Program
- NRC Safety Research Program Budget
- Severe Accident Research Program
- Evolutionary Light Water Reactor Design Certification Issues
- Human Factors and Other Organizational Issues
- Reactor Pressure Vessel Embrittlement
- NRC Computer Codes and Their Documentation
- Severe Accident Risk Assessment - NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants"

We have provided reports to the Nuclear Regulatory Commission and the NRC staff on the above-mentioned matters. Copies of these reports are enclosed.

We expect to continue to review various elements of the NRC Safety Research Program and provide reports to the Commission as warranted.

Sincerely,

David A. Ward
Chairman

*Enclosures:

1. Report from Carlyle Michelson, ACRS Chairman, to Kenneth M. Carr, U.S. NRC Chairman, Subject: Containment Performance Improvement Program - Proposed Recommendations for MARK II, MARK III, Ice Condenser, and Dry Containments, March 13, 1990.
2. Report from Carlyle Michelson, ACRS Chairman, to Kenneth M. Carr, U.S. NRC Chairman, Subject: NRC Safety Research Program Budget, April 11, 1990.
3. Report from Carlyle Michelson, ACRS Chairman, to Kenneth M. Carr, U.S. NRC Chairman, Subject: Severe Accident Research Program, April 24, 1990.
4. Report from Carlyle Michelson, ACRS Chairman, to Kenneth M. Carr, U.S. NRC Chairman, Subject: Evolutionary Light Water Reactor Certification Issues and Their Relationship to Current Regulatory Requirements, April 26, 1990.
5. Report from Carlyle Michelson, ACRS Chairman, to Kenneth M. Carr, U.S. NRC Chairman, Subject: NRC Research on Organizational Factors, August 16, 1990.
6. Report from Carlyle Michelson, ACRS Chairman, to Kenneth M. Carr, U.S. NRC Chairman, Subject: Yankee Rowe Reactor Pressure Vessel Integrity, September 12, 1990.
7. Report from Carlyle Michelson, ACRS Chairman, to James M. Taylor, Executive Director for Operations, U.S. NRC, Subject: NRC Computer Codes and Their Documentation, October 11, 1990.
8. Report from Carlyle Michelson, ACRS Chairman, to Kenneth M. Carr, U.S. NRC Chairman, Subject: Review of NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," November 15, 1990.

---

*For Items 1 through 8, see NUREG-1125, Volume 12, 4/91.

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

January 15, 1991

The Honorable Thomas S. Foley
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 has reported to the Congress each year on the Safety Research Program of the Nuclear Regulatory Commission. In our December 18, 1986 letter to the Congress, we proposed to provide reports on specific issues rather than one all-inclusive annual report.

During the past year, we have reviewed the NRC Safety Research Program and other closely related matters in the following areas:

- Nuclear Power Plant Containment Performance Improvement Program
- NRC Safety Research Program Budget
- Severe Accident Research Program
- Evolutionary Light Water Reactor Design Certification Issues
- Human Factors and Other Organizational Issues
- Reactor Pressure Vessel Embrittlement
- NRC Computer Codes and Their Documentation
- Severe Accident Risk Assessment - NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants"

We have provided reports to the Nuclear Regulatory Commission and the NRC staff on the above-mentioned matters. Copies of these reports are enclosed.

We expect to continue to review various elements of the NRC Safety Research Program and provide reports to the Commission as warranted.

Sincerely,

David A. Ward
Chairman

*Enclosures:

1. Report from Carlyle Michelson, ACRS Chairman, to Kenneth M. Carr, U.S. NRC Chairman, Subject: Containment Performance Improvement Program - Proposed Recommendations for MARK II, MARK III, Ice Condenser, and Dry Containments, March 13, 1990.
2. Report from Carlyle Michelson, ACRS Chairman, to Kenneth M. Carr, U.S. NRC Chairman, Subject: NRC Safety Research Program Budget, April 11, 1990.
3. Report from Carlyle Michelson, ACRS Chairman, to Kenneth M. Carr, U.S. NRC Chairman, Subject: Severe Accident Research Program, April 24, 1990.
4. Report from Carlyle Michelson, ACRS Chairman, to Kenneth M. Carr, U.S. NRC Chairman, Subject: Evolutionary Light Water Reactor Certification Issues and Their Relationship to Current Regulatory Requirements, April 26, 1990.
5. Report from Carlyle Michelson, ACRS Chairman, to Kenneth M. Carr, U.S. NRC Chairman, Subject: NRC Research on Organizational Factors, August 16, 1990.
6. Report from Carlyle Michelson, ACRS Chairman, to Kenneth M. Carr, U.S. NRC Chairman, Subject: Yankee Rowe Reactor Pressure Vessel Integrity, September 12, 1990.
7. Report from Carlyle Michelson, ACRS Chairman, to James M. Taylor, Executive Director for Operations, U.S. NRC, Subject: NRC Computer Codes and Their Documentation, October 11, 1990.
8. Report from Carlyle Michelson, ACRS Chairman, to Kenneth M. Carr, U.S. NRC Chairman, Subject: Review of NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," November 15, 1990.

---

*For Items 1 through 8, see NUREG-1125, Volume 12, 4/91.

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

February 12, 1991

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

Dear Mr. Taylor:

SUBJECT: STANDARD REVIEW PLAN FOR REVIEWING SAFETY ANALYSIS REPORTS FOR DRY METALLIC SPENT FUEL STORAGE CASKS

During the 370th meeting of the Advisory Committee on Reactor Safeguards, February 7-9, 1991, we considered a proposed Standard Review Plan (SRP) for Reviewing Safety Analysis Reports for Dry Metallic Spent Fuel Storage Casks. Our Subcommittee on Defueling and Fuel Pool Storage discussed this matter with the staff during a meeting on January 29, 1991. During our review we also had the benefit of the documents referenced.

The staff proposes to publish this document as a NUREG. We concur that it will provide useful guidance to those reviewing cask designs and to those who may submit designs for approval. We have the following comments:

1. The proposed SRP is a careful, thorough, and detailed description of a plausible review process. We did not identify any important safety question that was not explored.

2. The relationship of the proposed SRP to Regulatory Guide 3.61, "Standard Format and Content for a Topical Safety Analysis Report for a Spent Fuel Dry Storage Cask," appears to be unusual. The content of this regulatory guide more nearly resembles design criteria found in Appendix A to 10 CFR Part 50 than it does a typical regulatory guide. A typical regulatory guide gives much more specific guidance than does Regulatory Guide 3.61. Indeed, the proposed SRP resembles many existing regulatory guides more than Regulatory Guide 3.61 does. However, since applicants will have access to the SRP, perhaps it can serve as both a regulatory guide and a standard review plan.

3. In some areas the proposed SRP appears to be overly conservative. For example, the reviewer is to give no "credit for burnup nor the presence of neutron poisons formed during irradiation" in criticality calculations (p. 6-3). Thermal loading calculations are to "reflect the worst credible

combinations" of possible thermal loads (p. 4-2). Stress cycles due to "periodic precipitation of snow and possible formation of ice" are to be considered (p. 2-7). "The long term effect of these stress cycles should be addressed ..." (p. 2-7). For accident conditions "instantaneous release of 100 percent of the gaseous inventory should be assumed" (p. 7-4). There are others, but these are representative examples. We recommend language in the proposed SRP that encourages reviewer flexibility in considering alternatives in these areas.

4. In the version we examined there are some statements that would benefit from clarification. These statements were identified to the staff in the course of our review.

Sincerely,

David A. Ward

David A. Ward
Chairman

<u>References</u>:

1. U.S. Nuclear Regulatory Commission, Proposed NUREG, "Standard Review Plan for Reviewing Safety Analysis Reports for Dry Metallic Spent Fuel Storage Casks," transmitted by memorandum dated September 6, 1990 from John P. Roberts, NMSS, to William Kerr, ACRS
2. U.S. Nuclear Regulatory Commission, Regulatory Guide 3.61, "Standard Format and Content for a Topical Safety Analysis Report for a Spent Fuel Dry Storage Cask," February 1989

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

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

1. REPORT NUMBER (Assigned by NRC. Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG-1125
Volume 13

2. TITLE AND SUBTITLE

A Compilation of Reports of the Advisory Committee on Reactor Safeguards: 1991 Annual

3. DATE REPORT PUBLISHED

MONTH	YEAR
April	1992

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

6. TYPE OF REPORT

Compilation

7. PERIOD COVERED (Inclusive Dates)

Jan. thru Dec. 1991

8. PERFORMING ORGANIZATION – NAME AND ADDRESS (If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)

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

9. SPONSORING ORGANIZATION – NAME AND ADDRESS (If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)

Same as above

10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

This compilation contains 41 ACRS reports submitted to the Commission, Executive Director for Operations, or to the Office of Nuclear Regulatory Research, during calendar year 1991. It also includes a report to the Congress on the NRC Safety Research Program. All reports have been made available to the public through the NRC Public Document Room and the U. S. Library of Congress. The reports are divided into two groups: Part 1: ACRS Reports on Project Reviews, and Part 2: ACRS Reports on Generic Subjects. Part 1 contains ACRS reports alphabetized by project name and by chronological order within project name. Part 2 categorizes the reports by the most appropriate generic subject area and by chronological order within subject area.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)

Nuclear Reactors
Nuclear Reactor Safety
Reactor Operations
Safety Engineering
Safety Research

13. AVAILABILITY STATEMENT

Unlimited

14. SECURITY CLASSIFICATION

(This Page)
Unclassified

(This Report)
Unclassified

15. NUMBER OF PAGES

16. PRICE

NRC FORM 335 (2-89)

THIS DOCUMENT WAS PRINTED USING RECYCLED PAPER

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

OFFICIAL BUSINESS
PENALTY FOR PRIVATE USE, $300

SPECIAL FOURTH-CLASS RATE
POSTAGE AND FEES PAID
USNRC
PERMIT NO. G-67