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Volume 15

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

1993 Annual

U.S. Nuclear Regulatory
Commission

April 1994

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ABSTRACT

This compilation contains 47 ACRS reports submitted to the Commission, Executive Director for Operations, or to the Office of Nuclear Regulatory Research, during calendar year 1993. 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 categorized 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 1993. NUREG-1125 is published annually. Previous issues are as follows:

<u>Volume</u>	<u>Inclusive Dates</u>
1 through 6	September 1957 through December 1984
7	Calendar Year 1985
8	Calendar Year 1986
9	Calendar Year 1987
10	Calendar Year 1988
11	Calendar Year 1989
12	Calendar Year 1990
13	Calendar Year 1991
14	Calendar Year 1992

ACRS MEMBERSHIP (1993)

CHAIRMAN: Dr. Paul G. Shewmon, Professor Emeritus
Ohio State University
(Term ended 5/93)

VICE CHAIRMAN: Dr. J. Ernest Wilkins, Jr., Distinguished Professor
Clark Atlanta University
(Chairman by Succession 6/93)

MEMBERS: Mr. James C. Carroll, Retired
Pacific Gas & Electric Company
(Vice Chairman by Succession 6/93)

Dr. Ivan Catton, Professor
University of California, Los Angeles

Mr. Peter R. Davis, President
PRD Consulting, Idaho Falls

Dr. Thomas S. Kress
Oak Ridge National Laboratory

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

Mr. William J. Lindblad, Retired
Portland General Electric Company

Mr. Carlyle Michelson, Retired
Tennessee Valley Authority and
U. S. Nuclear Regulatory Commission

Dr. Robert L. Seale
University of Arizona

Dr. William J. Shack
Argonne National Laboratory

Mr. Charles J. Wylie, Retired
Duke Power Company

TABLE OF CONTENTS

	<u>Page</u>
ABSTRACT	iii
PREFACE	v
MEMBERSHIP	vii

Accidents and Incidents

SECY-92-413, "Incident Investigation Options Reporting to the Commission," January 13, 1993	1
--	---

Advanced Reactor Designs

Issues Pertaining to the Advanced Reactor (PRISM, MHTGR, and PIUS) and CANDU 3 Designs and Their Relationship to Current Regulatory Requirements, February 19, 1993	3
Advanced Boiling Water Reactor (ABWR) Review Schedule, March 18, 1993	9
SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," April 26, 1993	11
SECY-93-289, "Issuance of the Draft Preapplication Safety Evaluation Report (PSER) for the Power Reactor Innovative Small Module (PRISM) Liquid-Metal Reactor," November 10, 1993	15
NRC Confirmatory Test Program in Support of the AP600 Design Certification, November 18, 1993	17

TABLE OF CONTENTS

	<u>Page</u>
ACRS Review of the Advanced Boiling Water Reactor Final Safety Evaluation Report, December 15, 1993	21
Diversity in the Method of Measuring Reactor Pressure Vessel Water Level in the Advanced and Simplified Boiling Water Reactor Designs, December 16, 1993	23
Electric Power Research Institute Advanced Light Water Reactor Utility Requirements Document -- Volume III Passive Plants, December 23, 1993	25
<u>Electrical Power Systems</u>	
Proposed Generic Letter Regarding Removal of Accelerated Testing and Special Reporting Requirements for Emergency Diesel Generators from Plant Technical Specifications, September 22, 1993	29
NRC Staff Devotion to "Trigger Values" in Its Effort to Assure Emergency Diesel Generator (EDG) Reliability in the Context of the Rule on Station Blackout, December 14, 1993	33
<u>Emergency Planning</u>	
See "Regulatory Procedures"	105
<u>Fire Protection</u>	
Thermo-Lag Fire Barriers, December 16, 1993	37
See "Generic Issues/Unresolved Safety Issues"	47

TABLE OF CONTENTS

Page

Generic Issues/Unresolved Safety Issues

Proposed Resolution of Generic Safety Issue 120, "On-Line Testability of Protection Systems," January 13, 1993	39
Proposed Resolution of Generic Issue 142, "Leakage Through Electrical Isolators in Instrumentation Circuits," February 19, 1993	41
Prioritization of Generic Issue 152, "Design Basis for Valves That Might Be Subjected to Significant Blowdown Loads," April 23, 1993	43
Proposed Resolution of Generic Issue 105, "Interfacing Systems LOCA in LWRs," May 20, 1993	45
Proposed Resolution of Generic Issue 57, "Effects of Fire Protection System Actuation on Safety-Related Equipment," August 11, 1993	47
Proposed Resolution of Generic Issue 143, "Availability of Chilled Water System and Room Cooling," August 11, 1993	49
Proposed Priority Rankings of Generic Issues: Eighth Group, September 16, 1993	51

Human Factors

Human Performance in Operating Events, March 19, 1993	59
Review of Organizational Factors Research Program, April 27, 1993	63

TABLE OF CONTENTS

	<u>Page</u>
<u>Instrumentation, Control and Protection Systems</u>	
Computers in Nuclear Power Plant Operations, March 18, 1993	69
NRC/International Meetings on Digital Instrumentation and Control Matters - Russell's, NRR, Statement During NRC Staff's 5/14/93 Meeting with the Commission, June 18, 1993	75
Computers in Nuclear Power Plant Operations, November 16, 1993	77
<u>Materials Engineering</u>	
Proposed Supplement 6 to Generic Letter 89-10, "Information on Scope, Grouping, Prioritization, Schedule, and Public Questions," December 20, 1993	79
<u>Probabilistic Risk Assessment</u>	
Draft Report of the PRA Working Group, May 20, 1993	81
Draft Final Report of the PRA Working Group, November 10, 1993	83
<u>Reactor Operations</u>	
Staff Initiatives to Revise the Systematic Assessment of Licensee Performance Program, April 30, 1993	85
<u>Regulatory Guides</u>	
Proposed Final Versions of Regulatory Guides for Implementing Revised 10 CFR Part 20, "Standards for Protection Against Radiation," April 23, 1993	89

TABLE OF CONTENTS

Page

Proposed Draft Regulatory Guides, DG-1023, "Evaluation of Reactor Pressure Vessels With Charpy Upper-Shelf Energy Less Than 50 ft-lb," and DG-1025, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," July 15, 1993	91
See " Rules and Regulations "	119

Regulatory Procedures

Policy Statement on Technical Specifications Improvements for Nuclear Power Plants, June 18, 1993	93
Regulatory Review Group Report, July 15, 1993	97
Draft Commission Paper, "Policy and Technical Issues Associated With the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs," November 10, 1993	101
Draft Rulemaking Package Eliminating the Emergency Planning Annual Exercise, December 20, 1993	105

Rules and Regulations

SECY-93-049, Implementation of 10 CFR Part 54, Requirements for Renewal of Operating Licenses for Nuclear Power Plants, April 23, 1993	107
See " Regulatory Guides "	89
Backfit Rule, May 20, 1993	109

TABLE OF CONTENTS

	<u>Page</u>
SECY-93-113, Additional Implementation Information for 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," June 18, 1993	111
Implementation Guidance for the Maintenance Rule, April 26, 1993, Revised: June 24, 1993	113
See "Severe Accidents"	135
Proposed Rule Amending Fracture Toughness Requirements for Light Water Reactor Pressure Vessels, Proposed Rule Regarding Requirements for Thermal Annealing of Reactor Pressure Vessels, and Draft Regulatory Guide on Format and Content of Application for Approval for Thermal Annealing of Reactor Pressure Vessels, September 20, 1993	115
Proposed Final Amendments to 10 CFR Part 55 on Renewal of Licenses and Requalification Requirements for Licensed Operators, October 14, 1993	117
Proposed Rule and Draft Regulatory Guide to Address Resolution of Generic Issue 23, "Reactor Coolant Pump Seal Failure," October 14, 1993	119
Proposed Amendments to 10 CFR Part 73 to Protect Against Malevolent Use of Vehicles at Nuclear Power Plants, December 10, 1993	121
<u>Safety Philosophy, Technology & Criteria</u>	
Definition of a Large Release for Use With Safety Goal Policy, April 22, 1993	125

TABLE OF CONTENTS

Page

Staff Approach for Assessing the Consistency of the Present Regulations With Respect to the Commission's Safety Goals, May 26, 1993, Revised: June 16, 1993	127
---	-----

Safety Research

ACRS Report to Congress on NRC Safety Research Program, January 28, 1993	131
See " Human Factors "	63

Severe Accidents

Public Comments on Proposed Rule on ALWR Severe Accident Performance, June 18, 1993	135
Individual Plant Examination Program, December 16, 1993	137

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

January 13, 1993

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

Dear Chairman Selin:

SUBJECT: SECY-92-413, "INCIDENT INVESTIGATION OPTIONS REPORTING TO
THE COMMISSION"

During the 393rd meeting of the Advisory Committee on Reactor Safeguards, January 7-8, 1993, we discussed the staff's proposed options as described in SECY-92-413 for the Incident Investigation Program. The staff proposes the formation of an Incident Investigation Group (IIG). We generally endorse this proposal and would like to be kept informed of further progress in this area. However, we have some comments regarding options being considered for the makeup of the IIG membership.

As noted in SECY-92-413, it is important that the IIG be competent and independent in order to enhance its effectiveness and credibility. We believe that these attributes may be unnecessarily compromised by some of the proposed options. In particular, we are concerned about the exclusion of experts affiliated with the nuclear steam system suppliers or architect-engineers. These people should be among those who possess the highest level of expertise available. Further, we recommend that the participation in and control of the IIG by the NRC be minimized to help preserve at least the perception of independence.

We are in general agreement with the proposed purpose and scope of the IIG function. However, we urge that further consideration be given to allowing the IIG to make recommendations on the basis of its investigation. The staff's proposal recommends that the IIG report not contain recommendations.

Sincerely,

A handwritten signature in cursive script that reads "Paul Shewmon".

Paul Shewmon
Chairman

The Honorable Ivan Selin

2

January 13, 1993

Reference:

SECY-92-413, dated December 16, 1992, for the Commissioners from James M. Taylor, Executive Director for Operations, Subject: Incident Investigation Options Reporting to the Commission (Predecisional)

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

February 19, 1993

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

Dear Chairman Selin:

SUBJECT: ISSUES PERTAINING TO THE ADVANCED REACTOR (PRISM, MHTGR,
AND PIUS) AND CANDU 3 DESIGNS AND THEIR RELATIONSHIP TO
CURRENT REGULATORY REQUIREMENTS

During the 393rd and 394th meetings of the Advisory Committee on Reactor Safeguards, January 7-8 and February 11-13, 1993, we reviewed a draft Commission paper on the cited subject. Our Subcommittee on Advanced Reactor Designs also met on January 6, 1993, to discuss this matter. We had the benefit of discussions with representatives of the NRC staff, the Department of Energy, and the preapplicants: Atomic Energy of Canada, Limited, Technologies (AECLT), General Electric Nuclear Energy (GE), and General Atomics (GA). We also had the benefit of the referenced documents.

The draft Commission paper lists ten issues that need policy direction from the Commission for proposed deviations from existing regulations. These deviations arise either because existing regulations are generally specific to light water reactors (LWRs), or because the criteria proposed by the designers of the four reactor types listed are significantly different from those in the existing regulations. The draft paper also classified these ten issues into two categories: (1) those issues for which the staff agrees that departures from current regulations should be considered and (2) those issues for which the staff does not believe a departure from current regulations is warranted at this time. Not all of these issues are relevant to each reactor type; the draft paper contains a matrix identifying plant applicability. The paper contains some general comments and recommendations, as well as specific comments and recommendations on each of the ten issues.

Everything we say is predicated on our understanding of the applicable safety policies, which we would describe as follows:

- The safety objective for the nuclear enterprise was described in the 1986 Policy Statement on Safety Goals, and has not been rescinded. There is no distinction drawn in there between existing plants and new plants.

- The ACRS has recommended that the principal use of the goals be to judge the effectiveness of the entire enterprise, including regulation, in producing a plant population consistent with the goals. The Commission has never rejected that view.
- If the industry chooses to do better, we can only applaud its zeal, but ought not to stifle initiative by transforming initiatives into requirements.

Our views on the various items in the referenced draft paper are given below.

GENERAL COMMENTS

1. We find that the identified issues are important and that the staff should receive guidance from the Commission. (There are other policy issues affecting these reactor designs that are being addressed in connection with the evolutionary and passive LWR designs.) There may well be additional policy issues that appear during the preapplication review process. The staff has committed to identify any such issues in subsequent Commission papers.
2. The staff has grouped these ten issues into the two categories described above. We note that all of the affected preapplicants who appeared before us would treat Issue I (Control Room and Remote Shutdown Area Design) as a Category 1 issue, whereas the staff proposes it as a Category 2 issue. We will discuss this difference of opinion below in our opinion on Issue I.
3. For Category 1 issues, the staff proposes more conservative alternatives than the preapplicants propose, in order to account for uncertainties associated with the conceptual design. We are concerned that such an approach might well freeze an unnecessarily large degree of conservatism into the designs, and the preapplicants would have great difficulty persuading the staff to relax this conservatism on the basis of more precise information available in the final design.
4. We support the staff recommendation that "a prototype CANDU 3 is not required for design certification."
5. We support the staff intention to notify the Commission if its position on any of these ten issues should change, or if new issues are identified.

6. We have no objection to the staff recommendation that the highest priority be given to issues that are applicable to the PRISM design.
7. We understand and sympathize with the staff recommendation to defer decisions on generic rulemaking on these ten issues. Nevertheless, we urge the Commission to address these decisions in the near future. (The generic rulemaking question may arise in connection with passive LWR designs.)
8. In several places in the draft Commission paper, there occurs qualitative language, e.g., "appropriate conservatisms" or "credible severe accidents." This language must ultimately be translated into quantitative guidance. We believe that the quantitative guidance is, to a large measure, policymaking, and should not be relegated to low-level reviewers.

SPECIFIC COMMENTS

Category 1 Issues

A. Accident Evaluation

The staff proposal to develop a single approach with certain specified characteristics appears reasonable. We would like to review that approach when it is ready. We believe, however, that the staff should identify at an early stage quantitative guidelines and criteria for accident selection and evaluation. We note that AECLT has taken exception to some of the statements in the draft Commission paper that relate to its approach to this issue. We believe that this disagreement can be resolved by AECLT and the staff.

B. Source Term

The staff proposal to base the source terms on mechanistic analyses appears reasonable, although it is clear that the present data base will need to be expanded. We note that the staff is now developing for LWRs a revision to the TID-14844 source term. It will be appropriate for the staff to consider using the newer approach when it develops source terms, and to take specific account of the unique features of each of the reactor types.

C. Containment

The staff proposal "to postulate a core damage accident as a containment challenge ..." appears reasonable. We would like to review the list of postulated accidents when it is ready.

D. Emergency Planning

The staff proposes that advanced reactor licensees be required to develop offsite emergency plans which will include a requirement for onsite and offsite exercises. This proposal appears reasonable under the present circumstances, except that we would follow existing LWR guidance that permits the omission of offsite exercises when it can be shown that the design would preclude any accidental release exceeding the EPA Protective Action Guides. The staff has agreed to consider, after a review of Accident Evaluation (Issue A, above), whether some relaxation from current requirements may be appropriate. We urge that work on Issue D be closely correlated with work on Issues A and B, in order to avoid unnecessary conservatism.

E. Reactivity Control System

The staff proposal that the absence of control rods need not disqualify a reactor design, provided that an applicant can show a level of safety in reactor control equivalent to that of a traditional rodged system, appears reasonable. We note that this issue is applicable only to the PIUS concept, and that we have not yet had the benefit of presentations by the PIUS designers.

F. Operator Staffing and Function

The staff intends to review the justification for a smaller crew size by evaluating the function and task analyses for normal operation and accident management. This intention appears reasonable, although we believe that particular attention needs to be given to multiple module designs. We note that this issue is related to a similar issue for passive reactors. We believe that the Commission policy should be the same for the advanced reactors and CANDU 3 as it is for the passive reactors.

G. Residual Heat Removal

The staff belief that reliance on a single, completely passive, safety-related residual heat removal (RHR) system may be acceptable appears reasonable, although we would have liked to see the criteria to be used by the staff in deciding acceptability. We agree with the staff that NRC regulatory treatment of non-safety-related backup RHR systems for these reactors should be consistent with design requirements (not yet identified) for passive LWRs.

H. Positive Void Reactivity Coefficient

We agree with the staff that the existence of a positive void reactivity coefficient is a significant concern, but that it should not necessarily disqualify a reactor design. The burden of showing that the consequences of those accidents that would be aggravated by a positive void reactivity coefficient are either acceptable or could be satisfactorily mitigated by other design features surely falls on the preapplicant. On the other hand, the staff should state the criteria it will use to judge "acceptable" or "satisfactorily."

Category 2 Issues

I. Control Room and Remote Shutdown Area Design

We do not agree with the staff decision to treat this issue as a Category 2 issue, and the concomitant recommendation to apply current LWR regulations and guidance until passive LWR policy in this area is finalized. We believe that this issue should be a Category 1 issue, and that the preapplicants should accept the burden of convincing the staff that a proposed design is satisfactory, according to some criteria that should be specified by the staff.

J. Safety Classification of Structures, Systems, and Components

This issue is relevant only to the MHTGR concept. GA makes a persuasive case that the MHTGR is sufficiently different that the LWR criteria for identification of safety-related structures, systems, and components should not arbitrarily be applied to the MHTGR. We concur with this view and believe that Issue J should also be classified as a Category 1 issue. This would not preclude coordination of the policy for passive reactors with the policy for the MHTGR.

Our interest in all these matters continues. We would like an opportunity to review any significant change in staff or preapplicants position, as well as any significant developments in the implementation of the policies.

Dr. Thomas S. Kress did not participate in the Committee's deliberations regarding issues related to the MHTGR.

Sincerely,

Paul Shewmon
Chairman

References:

1. Memorandum dated December 2, 1992, from James M. Taylor, Executive Director for Operations, NRC, for the Commission, transmitting Advance Information Copy of Forthcoming Commission Paper - Issues Pertaining to the Advanced Reactor (PRISM, MHTGR, and PIUS) and CANDU 3 Designs and Their Relationship to Current Regulatory Requirements
2. Letter dated January 28, 1993, from David P. Hoffman, Gas-Cooled Reactor Associates, Management Committee, for D. M. Crutchfield, Office of Nuclear Reactor Regulation, NRC, Subject: Commission Papers on Policy Issues Concerning the Preapplication Reviews of Advanced Reactors
3. Letter dated January 25, 1993, from Peter M. Williams, Department of Energy, to J. Donohew, Office of Nuclear Reactor Regulation, NRC, commenting on the draft Commission Paper
4. Letter dated January 25, 1993, from N. Grossman, Department of Energy, to S. Sands, Office of Nuclear Reactor Regulation, NRC, Subject: Commission Papers on Policy Issues and Schedules Concerning the Preapplication Reviews of Advanced Reactor and CANDU 3 Designs

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

March 18, 1993

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

Dear Chairman Selin:

SUBJECT: ADVANCED BOILING WATER REACTOR (ABWR) REVIEW SCHEDULE

During the 395th meeting of the Advisory Committee on Reactor Safeguards, March 11-12, 1993, we discussed the staff's revised estimate of the schedule (proposed in SECY-93-041) for completing its review of the ABWR design. We also had the benefit of the documents referenced.

We note that in SECY-93-041, the time proposed for our review of the Final Safety Evaluation Report (FSER) is one month. In our July 18, 1991, report to you on "Schedules for Advanced Reactor Reviews," we agreed with the staff's estimate of three months for completing our review of the FSER. It is still our view that three months will be needed to perform a meaningful review, given the proposed schedule for transmitting the information to us.

Regarding our present ABWR review status, our work on the ABWR design certification application stalled in November 1992, pending the development of additional technical information by General Electric Nuclear Energy (GE) and decisions by the NRC staff on a number of important areas such as:

- design acceptance criteria/inspections, tests, analyses and acceptance criteria, digital control systems, control room and human factor provisions, and severe accident/probabilistic risk assessment considerations
- interface requirements and representative conceptual designs for uncertified portions of the design
- technical resolution of Unresolved Safety Issues and Generic Safety Issues as required by 10 CFR 52.47
- closure of open and confirmatory items in the October 1992 draft of the FSER

March 18, 1993

- closure of open items and concerns from the ACRS Advanced Boiling Water Reactors Subcommittee meetings of August 19, October 21, and November 18-19, 1992

Our subcommittee meetings with the NRC staff and GE were, in general, limited to consideration of the October 1992 draft of the FSER and the initial submittal and first twenty amendments (through March 13, 1992) of the ABWR Standard Safety Analysis Report (SSAR). We have not met with the staff or GE on these matters since November 1992, although we have planned a subcommittee meeting on severe accidents on March 18, 1993.

We will meet again to complete our review when the staff and GE provide us with reasonably complete final documentation for our consideration. There are now several additional voluminous amendments to the SSAR to consider, and extensive revision of the FSER is likely. From the nature of past ACRS open items and concerns on the ABWR and the uncertainty concerning their resolution, we believe that significant problems may still persist.

If it would expedite the schedule, we would be willing to meet with the staff and GE to review portions of the final FSER and associated SSAR beyond Amendment 20 as they are completed and made available. This would ensure a more timely resolution of any remaining concerns and could shorten the three months otherwise needed for our review of the advance copy of the complete FSER package (referred to in SECY-93-041) and preparation of our final report required by 10 CFR 52.53.

Sincerely,

Paul Shewmon
Chairman

References:

1. Letter dated February 9, 1993, from Dennis M. Crutchfield, NRR, to Paul Shewmon, Chairman, ACRS, Subject: Review Schedule for the Advanced Boiling Water Reactor (ABWR)
2. SECY-93-041, dated February 18, 1993, for the Commissioners from James M. Taylor, Executive Director for Operations, Subject: Advanced Boiling Water Reactor (ABWR) Review Schedule

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

April 26, 1993

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

Dear Chairman Selin:

SUBJECT: SECY-93-087, "POLICY, TECHNICAL, AND LICENSING ISSUES
PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER
REACTOR (ALWR) DESIGNS"

During the 396th meeting of the Advisory Committee on Reactor Safeguards, April 15-17, 1993, we discussed the NRC staff positions, delineated in SECY-93-087, on policy, technical, and licensing issues pertaining to evolutionary and advanced light-water reactor designs. During this meeting, we had the benefit of discussions with representatives of the NRC staff and of the documents referenced. We have discussed these issues during several of our previous meetings and provided comments and recommendations in the reports referenced.

We are in general agreement with the staff's positions in SECY-93-087; however, we have concerns regarding some issues and offer our comments and recommendations as follows. (The section titles and letter designations correspond to those in SECY-93-087.)

I. SECY-90-016 ISSUES

E. Fire Protection

In our April 26, 1990 report, we pointed out that redundant train separation is likely to be the most significant feature leading to reduced fire risk. We recommended that the proposed fire protection enhancements include separation of environmental control systems (i.e., separate heating, ventilating, and air conditioning (HVAC) systems for each train). The staff responded by conceding that separate HVAC arrangements may be needed, although other options may be available to the designer. The Commission endorsed the staff's response.

We remain concerned that a common normal ventilation system (such as that proposed for the ABWR) will be difficult to design to prevent the effluent from a postulated accident in one train of engineered safety features from reaching essential mitigating equipment in the other trains and

creating conditions that exceed their environmental qualifications. Of particular concern is the capability of ventilation dampers to isolate the effects of high energy pipe ruptures in confined compartments served by the common HVAC system.

G. Hydrogen Control

The staff claims that it has sufficient basis for understanding hydrogen behavior to go forward with licensing criteria. It has not been demonstrated to us that this basis is as extensive, or applicable, as the staff believes. Further, the AP600 and ABB-CE System 80+ designs have containments that are more susceptible to significant damage from hydrogen detonation than most existing and evolutionary plants. This requires that the licensing criteria for this issue be reconsidered.

H. Core Debris Coolability

The staff has weakened the position taken in SECY-90-016 by not requiring that the core debris be adequately quenched. We believe that the present criterion for coolability, namely a cavity floor area greater than $0.02\text{m}^2/\text{MWt}$, is not soundly based. We recommend that the staff validate containment response to core-on-the-floor accident sequences by independent analyses using, for example, MELCOR, or CORCON and CONTAIN.

J. Containment Performance

We agree with the requirement that containment stresses not exceed ASME Code Service Level C for metal containments, but it is not clear how electrical penetrations through the containment should be considered. Such penetrations utilize nonmetallic electrical insulation as a portion of the containment boundary and need further consideration.

L. Equipment Survivability

We agree that passive plant design features provided only for severe accident mitigation need not be subject to the environmental qualification requirements of 10 CFR 50.45. We believe, however, that such mitigation features must be designed to provide reasonable assurance that they will operate in the severe accident environment for which they are intended and over the timespan for which they are needed.

II. OTHER EVOLUTIONARY AND PASSIVE DESIGN ISSUES

Q. Defense Against Common-Mode Failure in Digital Instrumentation and Control Systems

The staff's second recommendation is that the vendor or applicant analyze each postulated common-mode failure for each event that is evaluated in the accident analysis section of the safety analysis report (SAR). We recommend that the scope of this assessment include consideration of common-mode failures during all events postulated in the SAR (e.g., fire, flood, pipe rupture, and extensive loss of essential power sources) and not be restricted to those events discussed in Chapter 15, "Accident Analysis."

T. Control Room Annunciator (Alarm) Reliability

The staff's basic recommendation is that the Commission approve the position that the alarm system for ALWRs meet the applicable EPRI requirements for redundancy, independence, and separation. These requirements do not include the use of Class 1E equipment and circuits. The staff also seeks approval of an additional position that goes beyond the EPRI requirements. This position is that "alarms that are provided for manually controlled actions for which no automatic control is provided and that are required for the safety systems to accomplish their safety functions, shall meet the applicable requirements for Class 1E equipment and circuits." We believe that the staff needs to provide clarification and additional justification for this position.

Collectively, our identified issues represent a significant array of incompletely addressed concerns. We urge that they be addressed on a timely basis to ensure their early consideration by the design teams.

Sincerely,

Paul Shewmon

Paul Shewmon
Chairman

References:

1. SECY-93-087, dated April 2, 1993, for the Commissioners, from James M. Taylor, Executive Director for Operations, NRC, Subject: Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactors (ALWR) Designs

2. Report from Paul Shewmon, ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: Computers in Nuclear Power Plant Operations, March 18, 1993
3. Report from David A. Ward, ACRS Chairman, to James M. Taylor, Executive Director for Operations, NRC, Subject: Draft Commission Paper, "Design Certification and Licensing Policy Issues Pertaining to Passive and Evolutionary Advanced Light Water Reactor Designs," September 16, 1992
4. Report from David A. Ward, ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: Digital Instrumentation and Control System Reliability, September 16, 1992
5. Report from David A. Ward, ACRS Chairman, to James M. Taylor, Executive Director for Operations, NRC, Subject: Issues Pertaining to Evolutionary and Passive Light Water Reactors and Their Relationship to Current Regulatory Requirements, August 17, 1992
6. Report from David A. Ward, ACRS Chairman, to James M. Taylor, Executive Director for Operations, NRC, Subject: Issues Pertaining to Evolutionary and Passive Light Water Reactors and Their Relationship to Current Regulatory Requirements, May 13, 1992
7. Report from Carlyle Michelson, ACRS Chairman, to Kenneth M. Carr, NRC Chairman, Subject: Evolutionary Light Water Reactors Certification Issues and Their Relationship to Current Regulatory Requirements, April 26, 1990

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

November 10, 1993

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

Dear Chairman Selin:

SUBJECT: SECY-93-289, "ISSUANCE OF THE DRAFT PREAPPLICATION SAFETY EVALUATION REPORT (PSER) FOR THE POWER REACTOR INNOVATIVE SMALL MODULE (PRISM) LIQUID-METAL REACTOR"

During the 403rd meeting of the Advisory Committee on Reactor Safeguards, November 4-6, 1993, we heard presentations by representatives of the NRC staff and General Electric Nuclear Energy on the subject SECY paper that proposes the issuance of a draft final Preapplication Safety Evaluation Report (PSER) for the Power Reactor Innovative Small Module (PRISM) Reactor for comment. We also had the benefit of the documents referenced.

Consistent with the Commission's advanced reactor policy, the staff has, to the extent feasible, used existing regulations to formulate criteria and procedures for review of this design. Where necessary the staff has created additional criteria and procedures, following the guidance furnished by the Commission in the Staff Requirements Memorandum dated July 30, 1993, that dealt with key policy issues for advanced reactors. Because the staff review was based on a conceptual design, the PSER did not, nor was it intended to, result in an approval of the design. Instead it identified certain key safety issues, provided some guidance on applicable licensing criteria, assessed the adequacy of the preapplicant's research and development programs, and concluded that no obvious impediments to licensing the PRISM design had been identified.

Although our own review of the PSER was less detailed than would have been appropriate for a safety evaluation report on an actual application, we believe that the staff has satisfactorily fulfilled its role in the preapplication process. We agree with the staff's proposal to provide the PSER to the U.S. Department of Energy.

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

Sincerely,

A handwritten signature in cursive script that reads "J. Ernest Wilkins, Jr." The signature is written in dark ink and is positioned above the printed name.

J. Ernest Wilkins, Jr.
Chairman

References:

1. SECY-93-289, dated October 19, 1993, Memorandum from James M. Taylor, NRC Executive Director for Operations, for the Commissioners, Subject: Issuance of the Draft Preapplication Safety Evaluation Report (PSER) for the Power Reactor Innovative Small Module (PRISM) Liquid-Metal Reactor
2. U.S. Nuclear Regulatory Commission, NUREG-1368, "Preapplication Safety Evaluation Report (PSER) for the Power Reactor Innovative Small Module (PRISM) Liquid Metal Reactor," October 1993
3. Staff Requirements Memorandum dated July 30, 1993, from S. Chilk, Secretary of the Commission, NRC, to J. M. Taylor, NRC Executive Director for Operations, Subject: SECY-93-092 - Issues Pertaining to the Advanced Reactor (PRISM, MHTGR, and PIUS) and CANDU 3 Designs and Their Relationship to Current Regulatory Requirements

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

November 18, 1993

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

Dear Chairman Selin:

SUBJECT: NRC CONFIRMATORY TEST PROGRAM IN SUPPORT OF THE AP600
DESIGN CERTIFICATION

During the 403rd meeting of the Advisory Committee on Reactor Safeguards, November 4-6, 1993, we reviewed selected aspects of the NRC Office of Nuclear Regulatory Research (RES) experimental program to be conducted at the Japan Atomic Energy Research Institute's (JAERI's) Large-Scale Test Facility (LSTF) in support of the NRC design certification of the Westinghouse (W) AP600 passive plant. Our Subcommittee on Thermal Hydraulic Phenomena met on October 28, 1993, to review this matter. During this review, we had the benefit of discussions with representatives of the NRC staff. We also had the benefit of the documents referenced.

In a September 16, 1992 Staff Requirements Memorandum, the Commission requested that the ACRS review selected aspects of the ROSA-V test program prior to its initiation. Specifically, the Committee was asked to review the test matrix and the facility modifications and additions, including instrumentation and controls. The following comments are offered in response to that request:

- The modified LSTF has been designated as ROSA-V. Despite the modifications, a number of atypicalities and scaling distortions exist in the ROSA-V configuration relative to the AP600 design. Some of the atypicalities in ROSA-V are: the use of one cold-leg per reactor coolant system (RCS) loop instead of two; the geometry and heat transfer characteristics of the steam generators; the existence of a four foot loop seal in the RCS; excess metal mass (in particular, for the core makeup tank (CMT)); the volume and geometry of the in-containment refueling water storage tank (IRWST); the primary residual heat removal (PRHR) system; and the configuration of the pressurizer surge line. RES staff representatives stated that they understand the impact these atypicalities will have on system performance. The RES staff has not, however,

presented a convincing argument that it understood the impact. RES should do so and document the results.

- Despite the facility shortcomings, we believe that ROSA-V will generate useful data to support validation of the relevant computer codes. This validation, however, may be inconclusive given the above atypicalities, especially those existing in the CMT, the PRHR system, and the IRWST. We recommend that the staff be urged to resolve the issues resulting from the atypicalities discussed above by additional analyses and, if necessary, by separate effects tests.
- The instrumentation proposed in support of the planned test program appears adequate for code assessment when dealing with single-phase phenomena. It is not clear that it is adequate for the measurement of key phenomena under conditions of two-phase flow. It is inadequate for determining some of the heat transfer characteristics of the PRHR system.
- The AP600 automatic depressurization system (ADS) will be activated by decreasing water level in the CMT. This level will be measured with heated junction thermocouples (HJTCs). The three AP600 integral system test facilities (ROSA-V, APEX-Advanced Plant Experiment-and SPES-II) will use differential pressure (DP) cells to measure this level. Activation of the ADS using DP cells rather than HJTCs could result in significant test distortions, given the inherent time delay associated with the use of HJTCs. The RES staff believes that these differences can be addressed. We were told by RES that JAERI has installed HJTCs of its own design at ROSA-V. We recommend that the RES staff use these HJTCs for ADS control for at least one properly chosen test, even if they are of a different design from those planned for use on the AP600.
- The ROSA-V test matrix is based on examination of transients and design-basis accidents for existing PWR designs. A number of the tests in the ROSA-V Phase I matrix have counterparts in the test matrices of the W SPES II and APEX facilities. These three facilities are scaled differently and have atypicalities of differing natures. We believe that the data obtained from these facilities will prove adequate for the necessary computer code validation by providing a broad range of challenges for simulation, given that the separate effects test programs supply sufficient information for code model development.
- Recently, RES modified the Phase I test matrix in response to a request from NRR to include some very small breaks and some "beyond-DBA" type events. We support this modification, but note that the capability of the relevant computer codes to

November 18, 1993

model very small-break LOCAs is weak. This may lead to difficulties when code validation is attempted.

Sincerely,

J. Ernest Wilkins, Jr.
Chairman

References:

1. U.S. NRC Report, NUREG/CR-6066 (Draft), "Analysis of LSTF Scaling for AP600 Testing," M. Ortiz, et al., June 11, 1993 (Draft Predecisional)
2. Memorandum dated December 23, 1992, from G. Rhee, NRC, to P. Boehnert, ACRS, transmitting INEL Report by T. Boucher, et al., "Description of Design Requirements for ROSA Modifications to Simulate AP600 Phenomena" (Revised September 1992)
3. U.S. NRC Report, NUREG/CR-5853, "Investigation of the Applicability and Limitations of the ROSA Large-Scale Test Facility for AP600 Safety Assessment," M. G. Ortiz, et al., December 1992
4. ACRS report dated July 17, 1993, "Integral System and Separate Effects Testing in Support of the Westinghouse AP600 Plant Design Certification"
5. Staff Requirements Memorandum dated September 16, 1992, from S. J. Chilk, Office of the Secretary, to J. M. Taylor, EDO, "SECY-92-219 - NRC-Sponsored Confirmatory Testing of the Westinghouse AP600 Design"
6. SECY-92-219, Memorandum dated June 16, 1992, from J. M. Taylor, NRC Executive Director for Operations, for the Commissioners, Subject: NRC-Sponsored Confirmatory Testing of the Westinghouse AP600 Design

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

December 15, 1993

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

Dear Chairman Selin:

SUBJECT: ACRS REVIEW OF THE ADVANCED BOILING WATER REACTOR FINAL
SAFETY EVALUATION REPORT

During the 404th meeting of the Advisory Committee on Reactor Safeguards, December 9-11, 1993, we discussed the schedule for completing our review of the NRC staff Final Safety Evaluation Report (FSER) for the General Electric Nuclear Energy (GENE) Advanced Boiling Water Reactor (ABWR) Standard Safety Analysis Report (SSAR). Previous schedules for our review of the ABWR were discussed in the referenced documents.

Our review of the FSER for the ABWR started with an ABWR Subcommittee meeting in October 1993, followed by another meeting in November. (During earlier Subcommittee meetings going back to 1989, we had reviewed ABWR/SER material.) Additional meetings are planned for December and January as advance copies of final draft material become available. Our Subcommittees on Computers in Nuclear Power Plant Operations, Design Acceptance Criteria, Severe Accidents, and Probabilistic Risk Assessment have met to review FSER areas of special interest to them.

The version of the FSER that we are reviewing is thought to cover most GENE submittals through Amendment 31 of the SSAR. This amendment was a reissuance of the complete SSAR in July 1993. Since then, GENE has issued an extensive revision as Amendment 32 and has just issued Amendment 33 on December 7, 1993. The staff intends to update its FSER through Amendment 33 during January 1994.

It appears likely to us that an additional SSAR amendment (beyond 33) will be needed to take care of a significant number of items that we have brought to the attention of GENE during and since our previous reviews of the SSAR (which were based on various earlier amendments). These items include numerous errors and inconsistencies in the SSAR and the absence of certain key information that we believe will be essential to obtaining a favorable Committee report. Some of these items were accommodated in Amendment 32.

Items brought to the attention of GENE by late November might be covered in Amendment 33. Additional items are likely to surface during the December and January Subcommittee meetings. All of our items must be closed with a final amendment issued by mid-February, reviewed expeditiously by the NRC staff, and considered by our ABWR Subcommittee at a meeting scheduled for March 9, 1994. We intend to complete our review and issue a final report only after the FSER is revised to reflect the final amendment to the SSAR.

On this basis, our ABWR Subcommittee will prepare, for full Committee consideration in March, a draft report on those portions of the ABWR application which concern safety. Barring untimely receipt of needed information or completion of the FSER revision, we expect to issue a final report to you in April 1994.

Sincerely,

J. Ernest Wilkins, Jr.
Chairman

References:

1. SECY-93-097, dated April 14, 1993, for the Commissioners from James M. Taylor, NRC Executive Director for Operations, Subject: Integrated Review Schedules for the Evolutionary and Advanced Light Water Reactor Projects
2. SECY-93-041, dated February 18, 1993, for the Commissioners from James M. Taylor, NRC Executive Director for Operations, Subject: Advanced Boiling Water Reactor (ABWR) Review Schedule
3. ACRS Report dated March 18, 1993, to Chairman Selin from Paul Shewmon, ACRS Chairman, Subject: Advanced Boiling Water Reactor (ABWR) Review Schedule

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

December 16, 1993

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

Dear Mr. Taylor:

SUBJECT: DIVERSITY IN THE METHOD OF MEASURING REACTOR PRESSURE
VESSEL WATER LEVEL IN THE ADVANCED AND SIMPLIFIED BOILING
WATER REACTOR DESIGNS

During the 404th meeting of the Advisory Committee on Reactor Safeguards, December 9-11, 1993, we discussed a proposal, advanced by representatives of the NRC staff, that General Electric Nuclear Energy (GENE) be required to install reactor pressure vessel (RPV) water level instrumentation that is diverse in operation from that presently employed on the Advanced Boiling Water Reactor (ABWR) and Simplified Boiling Water Reactor (SBWR) designs. During this meeting, we had the benefit of discussions with representatives of the NRC staff and GENE. We also had the benefit of the referenced documents.

We heard opposing views from the staff and GENE on the need for diversity in the method of measuring RPV water level in the ABWR and SBWR. The staff argues that "... two independent and diverse methods for measuring the RPV level should be required because of the importance of RPV level instrumentation to BWRs and because operating experience has shown the potential for failure of redundant level instruments due to common cause." The argument given by GENE is that the ABWR water level instrumentation is rugged, simple, and highly redundant with no known remaining operational problems. GENE further argues that alternate vessel level measurement technologies are unqualified for this application.

The staff has concluded that the differential pressure level measurement system employed in current BWRs provides adequate indication of reactor vessel water level. The staff has also concluded that the presently proposed ABWR level instrumentation meets the minimum requirements of all applicable General Design Criteria. It is the staff's interpretation, however, that this proposed instrumentation may not be in compliance with the relevant post-TMI requirement as codified in 10 CFR 50.34(f).

We do not believe that a case has been made by the staff for a water level indication system in advanced BWRs that is different from that currently used in operating BWRs.

Additional comments by ACRS Members Ivan Catton and Thomas S. Kress are presented below.

Sincerely,

J. Ernest Wilkins, Jr.
Chairman

Additional Comments by ACRS Members Ivan Catton and Thomas S. Kress

We agree that the present method of measuring vessel water level is sufficient for adequate protection for BWRs and that it is not appropriate to backfit new diversity into existing plants. Nevertheless, an objective of advanced and passive plants is to provide a higher level of safety assurance. We believe that the availability of at least three alternative level measuring methods affords an opportunity to provide this higher level of assurance in this important area. We agree with the staff's recommendation that installation of diverse vessel level instrumentation be required for the ABWR and SBWR designs.

References:

1. Proposed Draft SECY Paper (undated), from James M. Taylor, EDO, for the Commissioners, Subject, Diversity in the Method of Measuring Reactor Pressure Vessel Level in Advanced Boiling Water Reactor and Simplified Boiling Water Reactor (Draft Predecisional)
2. Memorandum dated December 10, 1993, from P. Boehnert, ACRS, for ACRS Members, Subject: ACRS Review of Proposed Requirement for Diverse Vessel Water Level Instrumentation for ABWR/SBWR - Additional Information on Diverse Level Instrumentation for German and Swedish BWR Plants

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

December 23, 1993

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

Dear Chairman Selin:

SUBJECT: ELECTRIC POWER RESEARCH INSTITUTE ADVANCED LIGHT WATER
REACTOR UTILITY REQUIREMENTS DOCUMENT -- VOLUME III
PASSIVE PLANTS

During the 402nd meeting of the Advisory Committee on Reactor Safeguards, October 7-8, 1993, we reviewed the staff Final Safety Evaluation Report (FSER) for Volume III of the Electric Power Research Institute (EPRI) Advanced Light Water Reactor (ALWR) Utility Requirements Document (URD) for Passive Plants. Our Subcommittee on Improved Light Water Reactors held a meeting on October 6, 1993, to review this subject. Our final deliberations on this matter occurred during our 404th meeting, December 9-11, 1993. 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.

In the early 1980s, EPRI established the ALWR program to support the United States utility industry efforts to ensure a viable nuclear power generation option for the 1990s and beyond. The overall objective was to establish utility industry policy along with top-tier technical and operational criteria for evolutionary and passive plant designs that would facilitate standardization and combined licensing. The intent of the program was to resolve as many of the policy, technical, and licensing issues as could be identified before specific plant designs were to be submitted, or approved. The remaining specific detailed technical and operational issues were to be resolved during consideration of detailed design information on specific plant design submittals. The program was to ensure that future nuclear power plants would be safer, simpler, more robust with greater margins, more easily operated and maintained, and more certain of being constructed and licensed without delays. The approach was to use utility experience to establish design philosophy, produce design criteria and guidance to achieve the objective, and to address the policies and regulations of the NRC.

The EPRI ALWR URD is a compendium of technical requirements for the design, construction, and performance of ALWR nuclear power plants for the 1990s and beyond. The URD consists of three volumes:

- Volume I, "ALWR Policy and Summary of Top-Tier Requirements," is a management-level synopsis of the URD, including the design objectives and philosophy, the overall physical configuration and features of a future nuclear plant design, and the steps necessary to take the proposed ALWR design criteria beyond the conceptual design state to a completed, functioning power plant.
- Volume II, "ALWR Evolutionary Plant," consists of 13 chapters and contains utility design requirements for evolutionary nuclear power plants.
- Volume III, "ALWR Passive Plant," consists of 13 chapters and contains utility design requirements for passive nuclear power plants.

We have followed the development of the EPRI ALWR program from its inception and offered suggestions regarding safety improvements on several occasions. We discussed development of the EPRI URD program and the NRC staff reviews during numerous Subcommittee and full Committee meetings. We previously presented our comments to the Commission pertaining to the FSER for Volume II by our report of August 18, 1992.

Volume III is similar to Volume II and many chapters are identical except for the features, requirements, and those policy, technical, and licensing issues unique to the passive plants. Although the Standard Review Plan (SRP) was used by the staff as guidance, the level of detail in the URD did not permit a verification of adequacy. (The SRP was written to support the review of the final safety analysis reports on specific plant designs for which a significant amount of design and construction information is normally available.) The staff conducted its review with the understanding that the EPRI design criteria would meet all current regulations, except where deviations were identified. The staff review of the URD focused on determining whether the EPRI criteria conflict with current regulatory requirements.

In addition, the staff identified a number of policy, technical, and licensing issues which needed resolution in order to complete its review of the ALWRs, including the URD. We provided comments on these issues by our referenced letters. The Commission considered the staff positions on twenty-one of the issues identified in SECY-93-087 pertaining to passive plants.

We believe that the staff has conducted a thorough and comprehensive review. We are in general agreement with the FSER pertaining

December 23, 1993

to Volume III and its conclusion that meeting the URD requirements could result in a reactor design that would not conflict with regulatory guidelines, and that would be responsive to various policy statements. Nevertheless, we are disappointed in the limited technical basis provided for several of the requirements relating to severe accidents - in particular hydrogen control, melt spreading and coolability, and fuel coolant interaction (steam explosion). In addition, we believe additional consideration should have been given to general design criteria for containment to withstand severe accident loads.

Sincerely,

J. Ernest Wilkins, Jr.
Chairman

References:

1. SECY-93-087, dated April 2, 1993, from James M. Taylor, Executive Director for Operations, for the Commissioners, Subject: Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs
2. SECY-92-172, dated May 12, 1992, from James M. Taylor, Executive Director for Operations, for the Commissioners, Subject: Final Safety Evaluation Report for Volume II of the Electric Power Research Institute's Advanced Light Water Reactor Requirements Document, including the following enclosures:
 - Draft Safety Evaluation Report for Volume I, "Program Summary of the NRC Review of the Electric Power Research Institute's Advanced Light Water Reactor Utility Requirements Document," prepared by the Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, dated May 1992
 - Final Safety Evaluation Report for Volume II, "NRC Review of the Electric Power Research Institute's Advanced Light Water Reactor Utility Requirements Document for Evolutionary Plant Designs," prepared by the Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, dated May 1992
3. Electric Power Research Institute, Advanced Light Water Reactor Utility Requirements Document, Volume II, "ALWR Evolutionary Plant," Chapters 1-13 through Revision 4, dated April 1992

4. Draft Commission Paper, undated, from James M. Taylor, Executive Director for Operations, for the Commissioners, Subject: Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs
5. Staff Requirements Memorandum dated July 21, 1993, from Samuel J. Chilk, Secretary, to James M. Taylor, Executive Director for Operations, Subject: SECY-93-087 - Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs
6. Letter dated November 10, 1993, from J. Ernest Wilkins, Jr., ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: Draft Commission Paper, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-safety Systems in Passive Plant Designs"
7. Letter dated April 26, 1993, from Paul Shewmon, ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs
8. Letter dated August 18, 1992, from David A. Ward, ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: Electric Power Research Institute Advanced Light Water Reactor Utility Requirements Document -- Volume II, Evolutionary Plants
9. Letter dated August 17, 1992, from David A. Ward, ACRS Chairman, to James M. Taylor, EDO, Subject: Issues Pertaining to Evolutionary and Passive Light-Water Reactors and Their Relationship to Current Regulatory Requirements
10. Letter dated May 13, 1992, from David A. Ward, ACRS Chairman, to James M. Taylor, EDO, Subject: Issues Pertaining to Evolutionary and Passive Light-Water Reactors and Their Relationship to Current Regulatory Requirements
11. Letter dated April 26, 1990, from Carlyle Michelson, ACRS Chairman, to Kenneth M. Carr, NRC Chairman, Subject: Evolutionary Light-Water Reactor Certification Issues and Their Relationship to Current Regulatory Requirements

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

September 22, 1993

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

Dear Chairman Selin:

SUBJECT: PROPOSED GENERIC LETTER REGARDING REMOVAL OF ACCELERATED
TESTING AND SPECIAL REPORTING REQUIREMENTS FOR EMERGENCY
DIESEL GENERATORS FROM PLANT TECHNICAL SPECIFICATIONS

During the 401st meeting of the Advisory Committee on Reactor Safeguards, September 9-10, 1993, we reviewed the subject generic letter (GL). During this meeting, we had the benefit of discussions with representatives of the NRC staff and NUMARC. We also had the benefit of the documents referenced.

The staff has informed us that this version of the proposed GL reflects consideration of the comments made by the Committee to Review Generic Requirements (CRGR). The proposed GL has been issued for public comment in accordance with our agreement that this could be done prior to our review.

The proposed GL would allow licensees to request removal of the Technical Specification (TS) provisions for accelerated testing and special reporting requirements for the emergency diesel generators (EDGs). When requesting this license amendment, licensees must, however, commit to implement a maintenance program for monitoring and maintaining EDG performance consistent with 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," and of Regulatory Guide (RG) 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," that was developed by the staff to provide guidance for complying with the provisions of the Maintenance Rule, 10 CFR 50.65.

In our April 26, 1993 (revised June 24, 1993) report on the draft version of RG 1.160, we noted that:

On many occasions, we have provided comments on the trigger-value approach proposed by the staff to resolve Generic Issue B-56, "Diesel Generator Reliability." The proposed regulatory guide for implementing the Maintenance Rule explicitly endorses the trigger-value

procedure for "monitoring emergency diesel generator (EDG) performance against EDG target reliability levels." It is categorically impossible to demonstrate the reliability of EDGs using this method. We remain strongly opposed to its use for this purpose and continue to recommend that the staff's implementation guidance for the Station Blackout Rule, 10 CFR 50.63, be revised to deal with this issue. When this is done, the regulatory guide should be appropriately revised.

The staff's response was to include a footnote in RG 1.160 which states:

The triggers are intended to indicate when emergency diesel generator performance problems exist such that additional monitoring or corrective action is necessary. It is recognized that it is not practical to demonstrate by statistical analysis that conformance to the trigger values will ensure the attainment of high reliability, with a reasonable degree of confidence, of individual EDG units.

We do not believe that this footnote satisfactorily resolves our concern.

Regulatory Guide 1.160 endorses Section 12.2.4 of NUMARC 93-01, "Industry Guidelines for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," which, in turn, references Appendix D of NUMARC 87-00, Revision 1, "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at LWRs." Each of these documents clearly implies that use of the "trigger values and monitoring methods" (as described in Appendix D of NUMARC 87-00, Revision 1) provides an acceptable means of monitoring EDG target reliabilities of 0.95 or 0.975 in accordance with the intent of 10 CFR 50.63 for coping with station blackouts. (See, for example, the language of the first paragraph of the introduction to Appendix D of NUMARC 87-00, Revision 1.) It can't be both ways! We strongly recommend that the staff and NUMARC collaborate in resolving this matter by appropriate revision of these documents.

We have had a longstanding concern that the EDGs at many nuclear power plants are being subjected to excessive and unnecessary surveillance testing and other testing as required by TS limiting conditions for operation, and that such testing may actually be degrading the reliability of these machines. Data for the years 1988 to 1991, provided to us by NUMARC, show that some EDGs are subjected to start testing only 12 to 15 times each year, while other EDGs are tested over 100 times each year.

This disparity in testing frequencies results, in part, from the wide variation in relevant TS requirements that were negotiated with licensees over the years. The fact that this situation has existed for so many years reflects badly on both the staff and licensees with respect to their effectiveness in dealing with an acknowledged problem having safety implications. We believe that this proposed GL is an important step in achieving a more rational testing program.

In addition to our recommendation that RG 1.160 and the NUMARC documents on which this proposed GL is based be revised to reflect statistical reality, we believe that the language of the proposed GL needs improvement. The proposed GL quotes a statement from RG 1.160 that triggers and testing of "problem diesels" will be addressed separately by the NRC. In the next paragraph of the GL, licensee commitments required for approval of the removal of accelerated testing and special reporting requirements from the TS are described, including the need for a commitment to RG 1.160. A statement is then made that these actions are intended to close the issues of triggers and testing for "problem diesels." The staff should clarify this apparent contradiction and state clearly that the former prescriptive requirement for accelerated testing has been eliminated by this proposed generic letter.

Sincerely,

J. Ernest Wilkins Jr.
Chairman

References:

1. Memorandum dated August 13, 1993, from J. Larkins, ACRS, to B. Grimes, Office of Nuclear Reactor Regulation, Subject: Proposed NRC Generic Letter "Removal of Accelerated Testing and Special Reporting Requirements for Emergency Diesel Generators from Plant Technical Specifications"
2. Memorandum dated August 12, 1993, from G. Marcus, Office of Nuclear Reactor Regulation, for J. Larkins, ACRS, forwarding proposed NRC Generic Letter Regarding Removal of Accelerated Testing and Special Reporting Requirements for Emergency Diesel Generators from Plant Technical Specifications
3. U.S. NRC Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," June 1993
4. NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," May 1993
5. Appendix D. "EDG Reliability Program" to NUMARC Report, "Guidelines and Technical Bases for NUMARC Initiatives"

- Addressing Station Blackout at LWRs," NUMARC 87-00, Revision 1, August 1991
6. SECY-93-044 dated February 22, 1993, for the Commission from James M. Taylor, NRC Executive Director for Operations, Subject: Resolution of Generic Safety Issue B-56, "Diesel Generator Reliability"
 7. Letter dated April 26, 1993 (Revised June 24, 1993), from Paul Shewmon, ACRS Chairman, to Brian K. Grimes, Office of Nuclear Reactor Regulation, Subject: Implementation Guidance for the Maintenance Rule
 8. Letter dated August 7, 1992, from Alex Marion, NUMARC, to Paul Boehnert, ACRS, providing Industrywide Emergency Diesel Generator Reliability Data
 9. Letter dated December 18, 1992, from Raymond Fraley, ACRS, to Alex Marion, NUMARC, Subject: Industrywide Emergency Diesel Generator Reliability Performance Data
 10. Letter dated March 1, 1993, from Alex Marion, NUMARC, to Raymond Fraley, ACRS, responding to questions regarding Industrywide Emergency Diesel Generator Reliability Performance Data

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

December 14, 1993

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

Dear Chairman Selin:

On several occasions we have written you about the staff devotion to "trigger values" in its effort to assure emergency diesel generator (EDG) reliability in the context of the rule on Station Blackout. We have said that the concept is statistically flawed, and in our last letter that it is categorically impossible to demonstrate the reliability of EDGs using these methods.

The attached response by the EDO seems to acknowledge the error, but states that the staff intends to make changes only in the Generic Letter, but not in the Regulatory Guide, because everyone knows the procedure is wrong. We find that a curious and unsatisfactory response. The EDO can doubtless outlast us, but that is hardly a proper remedy for mathematical error.

The EDO's response appears to suggest that the desire for mathematical rectitude is an unnecessary decoration in nuclear regulation. We disagree.

Sincerely,

J. Ernest Wilkins, Jr.
Chairman

Attachment:

Letter dated October 29, 1993, from James M. Taylor, EDO, to J. Ernest Wilkins, Jr., ACRS Chairman, regarding ACRS concern over "trigger value" approach proposed by Regulatory Guide 1.160

UNITED STATES
NUCLEAR REGULATORY COMMISSION

WASHINGTON, D.C. 20555-0001

October 29, 1993

J. Ernest Wilkins, Jr., Chairman
Advisory Committee on Reactor Safeguards
U. S. Nuclear Regulatory Commission
Washington, DC 20555

1993

Dear Chairman Wilkins:

In your letter to the Chairman, dated September 22, 1993, you expressed the Committee's concern regarding the "trigger-value" approach proposed by the staff in Regulatory Guide (RG) 1.160 to resolve Generic Issue B-56, "Diesel Generator Reliability." You stated that it is categorically impossible to demonstrate the reliability of emergency diesel generators (EDGs) using this method and that footnote 3 added to RG 1.160 does not satisfactorily resolve the Committee's concern. You also stated that the generic letter should be revised to remove the inconsistency regarding "problem diesel."

The staff agrees with the ACRS that conformance of individual EDG's with trigger values cannot be taken to mean, in any statistical fashion, that the EDG has demonstrated achievement of high reliability values such as 0.950 or 0.975, which licensees may have committed to when implementing the station blackout rule. Footnote 3 was added to RG 1.160 to emphasize this fact. If licensees choose to use trigger values, they can be used as alert levels to indicate EDG performance problems that may need additional monitoring or corrective actions. In addition to this prospective choice, RG 1.160 indicated that licensees also could use other performance measures such as establishing down-time constraints to ensure high EDG availability. NUMARC and individual licensees retain considerable flexibility in this matter and understand the proper usage of trigger values as set forth under the existing NUMARC maintenance guidance, 93-01 and RG 1.160. Trigger values as set forth in the NUMARC guidance and RG 1.160 do not invoke any requirements for accelerated testing. We continue to believe that NUMARC and individual licensees understand the proper usage of the trigger values as set forth in the NUMARC guidance and RG 1.160.

To avoid any misconceptions on the statistical significance of the trigger values and the related difficulties in implementation, the staff intends to remove reference to trigger values and the associated footnote from the generic letter. The staff's goal is for the licensee to maintain a high level of EDG reliability and availability. The staff believes that this goal can be achieved through implementation of the maintenance rule, which includes performance of a detailed root-cause analysis of all EDG failures, implementation of effective corrective actions taken in response to all failures, and implementation of an effective performance-based preventive

CONTACT: Om Chopra, EELB/DE
504-3265

J. Ernest Wilkins, Jr.

-2-

maintenance program. The generic letter will remain voluntary in nature for licensees to adopt and it is expected that those licensees most affected by requirements for accelerated EDG testing (and an inordinate amount of EDG failure reporting) will avail themselves of the revised technical specifications offered by the generic letter.

However, the staff will retain the requirement that licensees must provide a commitment to comply with the provisions of the maintenance rule and the guidance of RG 1.160 for emergency diesel generators as a necessary condition to eliminate the existing technical specification (TS) for the accelerated EDG testing and reporting requirements. We will also revise the generic letter to clarify that the need for the current prescriptive requirement in the plant TS for accelerated testing of a "problem diesel" has been eliminated with implementation of the maintenance rule.

We appreciate your comments and recommendations; however, we believe that it is unnecessary to revise RG 1.160 or ask NUMARC to revise any of the NUMARC documents to further clarify the issue of trigger values, which we believe is fully understood not to be a measure of reliability.

Sincerely,

James M. Taylor
Executive Director
for Operations

cc: The Chairman
Commissioner Rogers
Commissioner Remick
Commissioner de Planque
SECY
OGC
OCA
OPA

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

December 16, 1993

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

Dear Chairman Selin:

SUBJECT: THERMO-LAG FIRE BARRIERS

During the 404th meeting of the Advisory Committee on Reactor Safeguards, December 9-11, 1993, in response to the referenced Staff Requirements Memorandum, we discussed with representatives of the NRC staff, NUMARC, and industry the technical differences between NUMARC and the NRC staff on the NUMARC test program related to Thermo-Lag fire barriers. Our Subcommittee on Auxiliary and Secondary Systems discussed this matter during a meeting on November 19, 1993. We also had the benefit of the documents referenced.

At the beginning of our review of the Thermo-Lag fire barrier issue, there were several differences between the staff and NUMARC on how the tests should be instrumented and configured to demonstrate compliance with Appendix R. The differences were in the placement of the thermocouples, whether or not cables should be used in the cable trays during testing, and in post-test evaluation of the cable condition. NUMARC has now agreed to use the thermocouple placement suggested by the staff, and the staff appears to have agreed to some testing with cables in the cable tray. How the test results will be used remains open.

The principal concern of the staff is that the limited number of tests will not yield enough data for extrapolation to the large number of specific configurations needing evaluation. The difficulty is compounded by incomplete characterization of the thermophysical properties of Thermo-Lag. The data from the planned tests can be made much more broadly applicable by additional temperature measurements and engineering analysis. In particular, we recommend that the Thermo-Lag cold side surface temperature be measured and that several identical Thermo-Lag configurations be tested with different cable loadings, including no cable. The resulting data and analysis should allow plant-specific cabling and ampacity factors to be dealt with. It should also be possible to resolve NUMARC concerns about excessive conservatism.

December 16, 1993

Thermo-Lag provides protection from a fire, in part, by material ablation. This suggests to us that aged material may not perform as well as new material. We recommend that at least one test be duplicated with in-service aged Thermo-Lag.

Our interest in fire protection goes beyond the Thermo-Lag issue. We are concerned about the use of standards and practices that are based on fire protection standards developed for other industries. Their utilization for nuclear power plant application should be specifically evaluated. The move towards risk-based regulation leads us to question present fire risk methodologies, and the adequacy of fire science talent within the agency. We look forward to being kept informed by the staff and NUMARC when they reconsider current fire protection regulations.

Sincerely,

J. Ernest Wilkins, Jr.
Chairman

References:

1. Staff Requirements Memorandum, dated November 15, 1993, to J. M. Taylor, EDO, and J. T. Larkins, ACRS, from S. J. Chilk, Secretary, regarding the October 29, 1993 Commission Briefing on Thermo-Lag
2. Memorandum, dated November 10, 1993, to J. T. Larkins, ACRS, from A. Thadani, NRR, regarding ACRS Subcommittee Meeting on Thermo-Lag
3. Memorandum, dated October 8, 1993, for the Commissioners from J. M. Taylor, EDO, Subject: Quarterly Updates of the Thermo-Lag and Fire Protection Task Action Plans

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

January 13, 1993

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 120,
"ON-LINE TESTABILITY OF PROTECTION SYSTEMS"

During the 392nd and 393rd meetings of the Advisory Committee on Reactor Safeguards, December 9-11, 1992, and January 7-8, 1993, we discussed the NRC staff's actions and recommendations for resolution of Generic Safety Issue (GSI) 120, "On-Line Testability of Protection Systems." During these meetings, we had the benefit of discussions with representatives of the NRC staff, and of the documents referenced.

We agree with the NRC staff's conclusion that new regulatory requirements for existing plants are not justified, and the recommendation that GSI-120 be considered resolved. We recommend that the resolution also explicitly state how it applies to future plants.

Sincerely,

Paul Shewmon
Chairman

References:

1. Memorandum dated November 5, 1992, from W. Minners, Office of Nuclear Regulatory Research, NRC, for R. Fraley, ACRS, Subject: Recommended Resolution of GSI-120, "On-Line Testability of Protection Systems," w/Enclosures
2. U. S. Nuclear Regulatory Commission, NUREG/CR-5916, Subject: Technical Findings Related to Resolution of Generic Safety Issue 120, "On-Line Testability of Protection Systems," October 1992

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

February 19, 1993

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 142, "LEAKAGE
THROUGH ELECTRICAL ISOLATORS IN INSTRUMENTATION
CIRCUITS"

During the 394th meeting of the Advisory Committee on Reactor Safeguards, February 11-13, 1993, we discussed the resolution of the subject generic issue proposed by the Office of Nuclear Regulatory Research (RES). We had the benefit of the documents referenced.

We concur in the resolution of Generic Issue 142 proposed by RES and its recommendation that this Generic Issue be closed out. In the event that the resolution is changed from that which we considered, we wish to be informed and be given an opportunity for review prior to final disposition.

We also concur in the RES recommendation for development of guidance and revision to the Standard Review Plan for use in reviewing digital systems for future plants, or current plants where existing safety-related analog systems are replaced with digital systems. We wish to be kept informed and be given an opportunity for review as these are developed.

Sincerely,

Paul Shewmon
Chairman

References:

1. Memorandum, received February 10, 1993, from Warren Minners, Office of Nuclear Regulatory Research, NRC, for John T. Larkins, ACRS, Subject: Proposed Resolution of Generic Issue (GI) 142, "Leakage Through Electrical Isolators in Instrumentation Circuits," w/Attachments
2. U.S. Nuclear Regulatory Commission, NUREG/CR-5863, "Risk Assessment of Isolation Devices in Safety Systems," Sandia National Laboratories, January 1993

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

April 23, 1993

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

Dear Mr. Taylor:

SUBJECT: PRIORITIZATION OF GENERIC ISSUE 152, "DESIGN BASIS FOR VALVES THAT MIGHT BE SUBJECTED TO SIGNIFICANT BLOWDOWN LOADS"

During the 396th meeting of the Advisory Committee on Reactor Safeguards, April 15-17, 1993, we discussed the priority ranking of Generic Issue (GI) 152 proposed by the Office of Nuclear Regulatory Research (RES). Representatives of RES were available to answer questions and we had the benefit of the documents referenced.

The identification and prioritization of this issue were prompted by concerns expressed in our May 9, 1989, report to the Commission on Generic Letter (GL) 89-10 and our November 20, 1989, report to the Commission on GI-87. GL 89-10 deals with safety-related motor-operated valve testing and surveillance. GI-87 is related to high pressure coolant injection system and other motor-operated valves that may be required to close against high differential pressure and/or high flows associated with large pipe breaks.

In our November 20, 1989 report, we pointed out that the existing design basis for such valves in certain operating plants may not specify the "type of heavy duty" that is of concern (e.g., closure under pipe break conditions). If true, we were concerned that such unusual loading conditions might not be considered in the program required by GL 89-10 and that the deficiencies addressed by GI-87 would not be remedied.

Since it has been a long standing regulatory requirement to protect against postulated piping failures in fluid systems outside containment, it is likely that, by now, the staff has included our concern in the GL 89-10 program. If so, we recommend that GI-152 be withdrawn. If not, we would like to be informed.

Sincerely,

Paul Shewmon

Paul Shewmon
Chairman

References:

1. Memorandum dated January 22, 1993, from Eric S. Beckjord, Office of Nuclear Regulatory Research, for Warren Minners, Office of Nuclear Regulatory Research, Subject: Generic Issue No. 152, "Design Basis for Valves That Might be Subjected to Significant Blowdown Loads"
2. Report from Forrest J. Remick, ACRS Chairman, to Kenneth M. Carr, NRC Chairman, Subject: Proposed Resolution of Generic Issue 87 (GI-87), "HPCI Steam Line Break Without Isolation," November 20, 1989
3. Report from Forrest J. Remick, ACRS Chairman, to Lando W. Zech, Jr., NRC Chairman, Subject: Generic Letter on Safety-Related Motor-Operated Valve Testing and Surveillance, May 9, 1989

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

May 20, 1993

MEMORANDUM FOR: James M. Taylor
Executive Director for Operations

FROM: Paul Shewmon
Chairman

SUBJECT: PROPOSED RESOLUTION OF GENERIC ISSUE 105,
"INTERFACING SYSTEMS LOCA IN LWRs"

During the 397th meeting of the Advisory Committee on Reactor Safeguards, May 13-15, 1993, we reviewed the proposed resolution of Generic Issue 105, "Interfacing Systems LOCA in LWRs." During this meeting, we had the benefit of discussions with representatives of both the NRC Office of Nuclear Regulatory Research and its principal contractor (INEL). We also had the benefit of the referenced documents.

In our January 18, 1990, letter regarding this matter, we recommended that "Information developed by the staff in its interfacing systems LOCA program might be furnished to licensees for incorporation into their IPE programs...." The staff's proposed resolution is in essential agreement with this recommendation. We concur in the staff's proposed resolution of this issue.

Dr. J. Ernest Wilkins, Jr., did not participate in the Committee's deliberations regarding this matter.

Paul Shewmon

Paul Shewmon
Chairman

References:

1. Report dated January 18, 1990, from Carlyle Michelson, ACRS Chairman, to James M. Taylor, EDO, Subject: Resolution of the Interfacing Systems LOCA Issue
2. Memorandum dated April 2, 1993, from Warren Minners, Director, Office of Nuclear Regulatory Research, for John Larkins, ACRS, transmitting: (1) Draft Memorandum, from Eric S. Beckjord, Director, Office of Nuclear Regulatory Research, for James M. Taylor, EDO, Subject: Technical Resolution of Generic Issue 105 (GI-105) "Interfacing Systems Loss of Coolant Accident (ISLOCA) in LWRs"; (2) NRC Information Notice 93-XX:

Interfacing System LOCAs Outside Containment (GI-105), dated March 1993; (3) Draft NUREG-1463, dated March 1993, Subject: Regulatory Analysis for the Resolution of Generic Issue 105: Interfacing System Loss of Coolant Accident in Light Water Reactors; (4) Draft NUREG/CR-5928, EGG-2685, dated March 1993, Subject: ISLOCA Research Program Final Report, W. J. Galyean, D. L. Kelly, J. A. Schroeder, P. G. Ellison

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

August 11, 1993

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 57, "EFFECTS OF
FIRE PROTECTION SYSTEM ACTUATION ON SAFETY-RELATED
EQUIPMENT"

During the 400th meeting of the Advisory Committee on Reactor Safeguards, August 5-6, 1993, we reviewed the proposed Resolution of Generic Issue 57, "Effects of Fire Protection System Actuation on Safety-Related Equipment." During this meeting, we had the benefit of discussions with representatives of both the NRC staff and its contractor, Sandia National Laboratories. Our Subcommittee on Auxiliary and Secondary Systems reviewed this matter during a meeting on July 28, 1993. We also had the benefit of the documents referenced.

We concur with the NRC staff proposal to resolve Generic Issue 57 by allowing risk-significant contributors to be dealt with by the Individual Plant Examination of External Events (IPEEE, program).

In view, however, of the importance of fire and fire-related problems, we would like to meet with the staff and review a few selected IPEEEs to see how well this process of generic issue resolution is working. We would like to be sure that important risk contributors like fire are receiving the special attention they need and that appropriate changes in the plants are resulting.

Sincerely,

J. Ernest Wilkins, Jr.
Chairman

References:

1. SECY-93-143 dated May 21, 1993, for The Commissioners from James M. Taylor, Executive Director for Operations, NRC, Subject: NRC Staff Actions to Address the Recommendations in

- the Report on the Reassessment of the NRC Fire Protection Program
2. Sandia National Laboratories, NUREG/CR-5580, SAND90-1507, Vol. 1, Subject: Evaluation of Generic Issue 57: Effects of Fire Protection System Actuation on Safety-Related Equipment, December 1992
 3. U.S. Nuclear Regulatory Commission, Draft NUREG-1472, Subject: Regulatory Analysis for the Resolution of Generic Issue 57: Effects of Fire Protection System Actuation on Safety-Related Equipment, with a memorandum dated May 5, 1993, from Warren Minners, RES, for John T. Larkins, ACRS

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

August 11, 1993

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 143, "AVAILABILITY
OF CHILLED WATER SYSTEM AND ROOM COOLING"

During the 400th meeting of the Advisory Committee on Reactor Safeguards, August 5-6, 1993, we reviewed the NRC staff's proposed resolution of Generic Issue 143. During this meeting, we had the benefit of discussions with representatives of the NRC staff and its contractor, Pacific Northwest Laboratories. We also had the benefit of the documents referenced.

We agree with the staff's proposed resolution of Generic Issue 143. Although GI-143 has been resolved, we believe that the licensees should review this issue as part of their IPEs. We are concerned, however, that there appears to be no process in place to ensure that the licensees have given appropriate attention to this matter. We are also concerned about the large number of generic issues that have been dealt with in this manner, and we intend to address this matter in the near future.

Sincerely,

J. Ernest Wilkins, Jr.
Chairman

References:

1. Memorandum dated July 8, 1993, for John T. Larkins, ACRS, from Warren Minners, NRC/RES, transmitting Draft NUREG-1427, "Regulatory Analysis for the Resolution of Generic Issue 143, Availability of Chilled Water System and Room Cooling"
2. Pacific Northwest Laboratories Report, PNL-8750 (Draft), "Assessment of Affected Core Damage Frequency and Public Risks Associated with Generic Issue 143 - Availability of HVAC and Chilled Water Systems," April 1993

Mr. James M. Taylor

2

August 11, 1993

3. ACRS letter dated October 15, 1987, to V. Stello, EDO, from W. Kerr, ACRS Chairman, Subject: ACRS Comments on Nuclear Power Plant Air Cooling Systems

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

September 16, 1993

MEMORANDUM TO: James M. Taylor
Executive Director for Operations

FROM: J. Ernest Wilkins, Jr.
Chairman

SUBJECT: PROPOSED PRIORITY RANKINGS OF GENERIC ISSUES:
EIGHTH GROUP

During the 401st meeting of the Advisory Committee on Reactor Safeguards, September 9-10, 1993, we reviewed the priority rankings proposed by the NRC staff for the generic issues listed in the attached table. During this meeting, we had the benefit of discussions with representatives of the NRC staff.

We agree with all of the proposed priority rankings. We note, however, that the safety concerns of several generic issues are to be evaluated and resolved through the Individual Plant Examination (IPE) and Individual Plant Examination of External Events (IPEEE) processes, or are to be treated as licensing issues, or to be handled in some other manner. We are not sure whether the large number of licensees who have already completed their IPEs have evaluated these issues. We believe that the staff, while reviewing the IPE and IPEEE submittals, should ensure that these generic issues have been evaluated and resolved by the licensees in an adequate manner. In the event some of the licensees had not considered these issues in their IPEs or IPEEEs, the staff should establish a mechanism to require them to evaluate these issues. The staff should also establish a system to keep track of the resolution of all of the issues that have been included in other programs for resolution. We would like to hear a report from the staff, at an appropriate time, on the adequacy of resolution of all these generic issues.

A handwritten signature in cursive script, reading "J. Ernest Wilkins, Jr.", is positioned above the printed name.

J. Ernest Wilkins, Jr.
Chairman

Attachment:
As stated

GENERIC ISSUES REVIEWED BY THE ACRS
DURING THE SEPTEMBER 9-10, 1993 MEETING

Generic Issue Number	Title	Priority Ranking Proposed by the NRC Staff	Reference Document
2	Failure of Protective Devices on Essential Equipment	DROP	Memorandum from E. S. Beckjord for W. Minners, July 1, 1992
76	Instrumentation and Control Power Interactions	DROP (The safety concerns of this issue will be addressed more directly on a plant-specific basis in the IPE program)	Memorandum from E. S. Beckjord for W. Minners, April 20, 1992
78	Monitoring of Fatigue Transient Limits for Reactor Coolant System	MEDIUM	Memorandum from E. S. Beckjord for W. Minners, July 10, 1992
89	Stiff Pipe Clamps	LOW (For existing plants) MEDIUM (For future plants)	Memorandum from E. S. Beckjord for W. Minners, August 12, 1992
110	Equipment Protective Devices on Engineered Safety Features	DROP	Memorandum from E. S. Beckjord for W. Minners, June 5, 1992
132	RHR Pumps Inside Containment	DROP	Memorandum from E. S. Beckjord for W. Minners, March 31, 1992
144	SCRAM Without a Turbine/Generator Trip	LOW	Memorandum from E. S. Beckjord for W. Minners, March 12, 1992

Generic Issue Number	Title	Priority Ranking Proposed by the NRC Staff	Reference Document
145	Actions To Reduce Common Cause Failures	NEARLY RESOLVED (The IPE program has already requested licensees to consider common cause failures, and the regulatory guide being developed to implement the Maintenance Rule will propose monitoring failure rates to look for common cause)	Memorandum from E. S. Beckjord for W. Minners, February 11, 1992
147	Fire-Induced Alternate Shutdown/Control Room Panel Interactions	LICENSING ISSUE (The staff will develop guidance for review of the IPEEE submittals by the licensees)	Memorandum from E. S. Beckjord for W. Minners, August 26, 1992
148	Smoke Control and Manual Fire-Fighting Effectiveness	LICENSING ISSUE (The staff will develop guidance for review of the IPEEE submittals by the licensees)	Memorandum from E. S. Beckjord for W. Minners, August 26, 1992
155.1	More Realistic Source Term Assumptions	NEARLY RESOLVED (This issue is being pursued by the staff as part of the comprehensive revisions to 10 CFR Parts 50 and 100 to reflect a better understanding of accident source terms and severe accident insights. A replacement for TID-14844 is being formulated, based on recent severe accident research findings)	Memorandum from E. S. Beckjord for W. Minners, February 26, 1992

Generic Issue Number	Title	Priority Ranking Proposed by the NRC Staff	Reference Document
155.2	Establish Licensing Requirements for Non-Operating Facilities	REGULATORY IMPACT ISSUE	Memorandum from E. S. Beckjord for B. Morris, April 16, 1992
155.3	Improve Design Requirements for Nuclear Facilities	DROP (Of the four recommendations contained in this issue, two are being addressed in other ongoing programs and one has been addressed previously by the staff; the remaining recommendation has very little safety significance)	Memorandum from E. S. Beckjord for W. Minners, January 25, 1993
155.4	Improve Criticality Calculations	DROP (The safety concerns of this issue were addressed in the Severe Accident Research Program, Task 4.3: Investigate the Possibility and Consequences of Recriticality in Degraded BWR Cores)	Memorandum from E. S. Beckjord for W. Minners, August 14, 1992
155.5	More Realistic Severe Reactor Accident Scenario	DROP (The safety concerns of this issue are being addressed in the Severe Accident Research Program, Issue L2: In-Vessel Core Melt Progression and Hydrogen Generation)	Memorandum from E. S. Beckjord for W. Minners, June 15, 1992

Generic Issue Number	Title	Priority Ranking Proposed by the NRC Staff	Reference Document
155.6	Improve Decontamination Regulations	DROP (The safety concerns of this issue are being addressed in the ongoing rulemaking on Residual Contamination Criteria)	Memorandum from E. S. Beckjord for W. Minners, August 20, 1992
155.7	Improve Decommissioning Regulations	DROP (The safety concerns of this issue have already been addressed in several NRC programs)	Memorandum from E. S. Beckjord for W. Minners, April 20, 1992
156.1.1	Settlement of Foundations and Buried Equipment	DROP (The safety concerns of this issue will be addressed in the IPEEE program)	Memorandum from E. S. Beckjord for W. Minners, August 12, 1992
156.1.4	Industrial Hazards	DROP (The safety concerns of this issue will be addressed in the IPEEE program)	Memorandum from E. S. Beckjord for W. Minners, March 6, 1992
156.2.1	Severe Weather Effects on Structures	DROP (The safety concerns of this issue will be addressed in the IPEEE program)	Memorandum from E. S. Beckjord for W. Minners, January 29, 1992
156.2.2	Design Codes, Criteria, and Load Combinations	DROP (The safety concerns of this issue will be addressed in the IPEEE program)	Memorandum from E. S. Beckjord for W. Minners, July 9, 1992

Generic Issue Number	Title	Priority Ranking Proposed by the NRC Staff	Reference Document
156.2.3	Containment Design and Inspection	DROP (The safety concerns of this issue were addressed in the resolution of Generic Issue 118, "Tendon Anchor Head Failure")	Memorandum from E. S. Beckjord for W. Minners, May 4, 1992
156.2.4	Seismic Design of Structures, Systems, and Components	DROP (The safety concerns of this issue will be addressed in the IPEEE program)	Memorandum from E. S. Beckjord for W. Minners, March 6, 1992
156.3.1.1	Shutdown Systems	DROP (The safety concerns of this issue have been addressed in USI A-45 and the IPE and IPEEE programs)	Memorandum from E. S. Beckjord for W. Minners, March 10, 1992
156.3.1.2	Electrical Instrumentation and Controls	DROP (The safety concerns of this issue have been addressed in USI A-45 and the IPE and IPEEE programs)	Memorandum from E. S. Beckjord for W. Minners, March 10, 1992
156.3.2	Service and Cooling Water Systems	DROP (The safety concerns of this issue have been or are being addressed in other NRC programs)	Memorandum from E. S. Beckjord for W. Minners, March 11, 1992

Generic Issue Number	Title	Priority Ranking Proposed by the NRC Staff	Reference Document
156.3.3	Ventilation Systems	DROP (The safety concerns of this issue are either being addressed in staff actions on Generic Issues 83, 106, 136, 143, and 148, or are covered by existing regulations)	Memorandum from E. S. Beckjord for W. Minners, September 4, 1992
156.3.6.1	Emergency AC Power	DROP (The safety concerns of this issue have been or will be addressed in the resolution of USI A-44 and Generic Issues B-56 and 128)	Memorandum from E. S. Beckjord for W. Minners, February 25, 1992
156.3.6.2	Emergency DC Power	LOW	Memorandum from E. S. Beckjord for W. Minners, March 8, 1993
156.3.8	Shared Systems	DROP (The safety concerns of this issue have been addressed in USI A-44 and Generic Issues 43, 130, and 153)	Memorandum from E. S. Beckjord for W. Minners, June 15, 1992
156.4.2	Testing of the Reactor Protection System and the Engineered Safety Features	DROP (The safety concerns of this issue have been addressed in the resolution of Generic Issue 120)	Memorandum from E. S. Beckjord for W. Minners, March 10, 1992

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

March 19, 1993

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

Dear Chairman Selin:

SUBJECT: HUMAN PERFORMANCE IN OPERATING EVENTS

During the 391st meeting of the Advisory Committee on Reactor Safeguards, November 5-7, 1992, we discussed with representatives of the Office for Analysis and Evaluation of Operational Data (AEOD) a draft of the AEOD study entitled, "Operating Experience Feedback Report - Human Performance in Operating Events." (This study was issued as NUREG-1275 in December 1992.) Representatives of NUMARC provided comments on the draft of this study during our meeting. We also discussed this matter during our 395th meeting, March 11-12, 1993. We had the benefit of the documents referenced.

This study was conducted over a 2 1/2-year period and involved 16 onsite visits by multidisciplinary teams led by an AEOD staff member for the purpose of evaluating human performance during selected nuclear power plant events. The study focused on factors that influenced operator performance during a wide variety of plant events. AEOD estimates that these events represent approximately 30 percent of the events that challenged operating crews during this 2 1/2-year period. The study summarizes each event and the findings that the teams made, provides observations discerned from related events, and presents conclusions concerning overall human performance. These conclusions fall into four categories of human performance issues: control room organization, procedures, human-machine interface, and industry initiatives. Finally, the study attempts to compare the "latent factors" among these 16 events.

Five of the 16 events studied were also the subject of Augmented Inspection Teams (AITs). We believe that a number of the remaining 11 events were of sufficient significance from a human and organizational performance point of view to have warranted an AIT effort. During our meeting with the AEOD staff we commented that the final version of the study should address this issue, since it may be a weakness in the approach being used by the Office of Nuclear Reactor Regulation (NRR) and the Regional Offices in systematically analyzing and evaluating human performance in operating events. AEOD did not explicitly deal with this issue in the final version of the study.

We have been critical of AEOD in the past for its reluctance to discuss the performance of NRC staff organizations in the course of carrying out studies of this nature. It continues to be our view, as discussed under Summary and Conclusions below, that this should be a necessary part of AEOD studies of this nature.

The Analysis section of the study (Section 3.0) contains a number of observations and conclusions that we believe are of importance from a nuclear safety perspective. We have the following comments on this section of the study:

- Control room organizational weaknesses were observed in the response of some operating crews to emergency situations (Section 3.2). This matter should receive prompt attention by the staff, with appropriate involvement of NUMARC and/or the Institute of Nuclear Power Operations. The requisite organizational factors approaches needed to deal with emergency situations should be well understood at this stage of maturity of nuclear power plant operations. In addition to the lessons learned in response to actual emergency situations, the staff and licensees have had numerous opportunities to observe and learn from operating crew response during requalification examinations and emergency plan exercises. We recommend that the weaknesses observed be corrected on an expedited basis.
- We are concerned by the two events in which engineered safety features (ESFs) were bypassed (Section 3.3.4). (Neither of these events was raised to the level warranting an AIT and, in one of these cases, the ESF was bypassed without the knowledge of the shift supervisor.) It is not clear from the study if these events were investigated appropriately by the Regional Offices. We believe that occurrences of this kind may represent a serious "safety culture" problem within the licensee organization. The staff should thoroughly review licensee corrective actions for events of this nature to ensure that the real root causes of the events have been dealt with in a manner that will prevent their recurrence. We do not believe that it is sufficient for the licensee to state in its licensee event report (LER) that the control room operator was reprimanded and provided with remedial training; the licensee needs to thoroughly evaluate and correct any "safety culture" issues raised by such events. However, we caution against the staff assuming the role of "de facto management" by prescribing, as opposed to reviewing, licensee management actions.
- We are concerned by the statement in Section 3.5.1 that licensees had prepared an LER "in a most every case" but that "In some cases, it was difficult to tell that the reports (LERs) described the same event. It appears in these cases

March 19, 1993

that the licensee failed to consider the human performance aspects of the event or failed to include that information in the report." During our meeting with the staff, we suggested that the draft study would be strengthened by including a discussion of the completeness of each associated LER with the evaluation of the individual events. We also suggested that a more detailed evaluation be made of this apparent weakness in the present LER program. AEOD chose not to follow our suggestions.

Summary and Conclusions

We believe that the AEOD study has been useful in focusing the attention of NRR and the Regional Offices, as well as that of the industry, on human and organizational performance issues. We agree with AEOD's plan to continue this activity (as described in Section 4.0 of the study) until these issues have been effectively addressed. As discussed above, we recommend that the Commission provide policy direction to AEOD on the matter of its charter, with respect to evaluating the performance of NRC staff organizations in the course of carrying out studies of this nature.

Sincerely,

Paul Shewmon
Chairman

References:

1. U. S. Nuclear Regulatory Commission, NUREG-1275, Volume 8, "Operating Experience Feedback Report - Human Performance in Operating Events," December 1992
2. SECY-92-407, dated December 9, 1992, for the Commissioners from James M. Taylor, Executive Director for Operations, Subject: The Independent Role of the Office for Analysis and Evaluation of Operational Data in the Assessment of Operational Experience and the Investigation of Operational Events

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

April 27, 1993

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

Dear Chairman Selin:

SUBJECT: REVIEW OF ORGANIZATIONAL FACTORS RESEARCH PROGRAM

During the 392nd, 394th, 395th and 396th meetings of the Advisory Committee on Reactor Safeguards, December 9-11, 1992, February 11-13, March 11-12, and April 15-17, 1993, respectively, we discussed the Office of Nuclear Regulatory Research (RES) budget for the human factors research program and SECY-93-020, "Review of Organizational Factors Research." In addition, during our February 11-13, 1993 meeting, representatives of the NRC staff and two of the contractors involved in the organizational factors research program (Brookhaven National Laboratory and University of California at Los Angeles) discussed their work. (The other contractors are the Pennsylvania State University and the Accident Prevention Group, Inc.) We also had the benefit of the document referenced.

Members of our Human Factors Subcommittee and two subcommittee consultants attended the November 12, 1992, senior staff management workshop on the organizational factors research program.

ACRS has followed this program since it was revived in 1987. SECY-93-020 provides the results of the comprehensive review performed by RES of its organizational factors research program and a description of changes to be made to the program as a result of this review. In the Summary Section of this SECY document, RES concludes that

there is a relatively low cost-effectiveness in continuing regulatory research beyond FY 1993, until it is determined that organizational factors can be reliably integrated into PRA models. RES is meeting with NRR to coordinate further development of human reliability analysis modeling of organizational factors for PRA. It is possible that this further effort will continue at a low level of funding in FY 1994.

We were told that RES does not, at this time, propose to fund additional organizational factors research beyond FY 1993. We also learned from our discussions with RES representatives that its

Nuclear Safety Research Review Committee had not reviewed and provided comments on the need for continuing this program prior to the issuance of SECY-93-020.

After extensive deliberations, we have been unable to arrive at a consensus with respect to the continuation of this research activity. We plan to take this matter up again when NRR completes its user needs evaluation with respect to organizational factors research.

Additional comments by ACRS Members James C. Carroll, Ivan Catton, Peter R. Davis, and Robert L. Seale are presented below.

Sincerely,

Paul Shewmon
Chairman

Additional Comments by ACRS Members James C. Carroll, Ivan Catton, Peter R. Davis, and Robert L. Seale

We believe that the present organizational factors research effort should be continued to the point where a set of useful products becomes available for trial use by the staff and the nuclear utilities. Our reasons for this view are summarized below.

The Relationship Between Organizational Performance and Safety - The Historical Perspective Section of SECY-93-020 states that "poor organizational performance can be a major contributor to safety significant events and that there is a need for an improved technical base for determining the impact of organizational performance on safety." We agree and further believe that this is one of the most important safety issues presently facing the nuclear power industry. The industry knows how to design extremely safe plants from a hardware point of view. However, operating experience indicates that there are many outstanding questions with respect to the ability of the nuclear utilities in the U.S. (and worldwide) to safely manage the operation and maintenance of both operating and future nuclear power plants. The organizational performance of the NRC staff is also of concern to us in that it can have an impact on the safety of the regulated industry.

We note that the SECY paper describes the organizational factors research programs being carried out by the regulatory authorities in Sweden, the UK, and France. This raises the obvious question as to why RES has concluded that its program is not cost-effective while other nations' regulatory authorities are actively pursuing this issue. We believe that it is of interest that none of these

foreign programs are attempting to integrate organizational factors into PRAs.

It is our view that management science is a real and sophisticated academic field that needs to be tapped if the industry is to continue to make progress in dealing with organizational performance issues. There appears to be a lack of communication between the management science academic community and most policy-makers out in the "real world" of nuclear power plant regulation and operations. We believe that the Commission should encourage the involvement of the management science community in helping to improve the organizational performance of both the staff and the nuclear utilities.

RES Arguments for Terminating Organizational Factors Research - In SECY-93-020, RES makes the point that "the gathering of organizational factors data is resource intensive," but does not attempt to quantify this term. The presentations made to the Committee by the current contractors suggest that much less resource intensive approaches, relative to those used in the early phases of this work, are possible. The real test will be in the application of the products of this research when the benefits obtained can be compared to the resources invested.

RES also states that "there is a relatively low cost-effectiveness in continuing regulatory research beyond FY 1993, until it can be determined that organizational factors can be reliably integrated into PRA models." We were told by the contractors that the development and validation of these measurement tools are necessary before the integration of organizational factors into PRA models can be properly demonstrated. RES appears to have created a classic catch 22 situation in the position it has taken.

The Implications of Terminating Organizational Factors Research - RES states in the SECY paper that "the research products developed to date will be integrated by the end of FY 1993 for possible use in inspection and diagnostics evaluations." Based on our discussions with the contractors, we have concluded that the program to develop and verify organizational factors measurement tools is far from being completed. It appears to us that there is a major risk in exporting the present products to the field, since their almost certain unsuccessful application will bring this work into disrepute and create a significant obstacle to future developments in this field.

The Cost of Completing the Present Organizational Factors Research Program - The contractors were asked for their estimates of the time and cost to carry the present research to the point where a set of useful products (both organizational factors measurement tools and PRA modeling techniques) would become available for trial use by the staff and the nuclear utilities. They indicated that

this would require an additional three years of effort at an annual funding of about \$0.5 million (a small fraction of the current research program support budget). This additional \$1.5 million expenditure is to be contrasted with the \$3.8 million that has been expended on organizational factors research since 1987.

Our Reasons for Supporting Continuation of the Present Organizational Factors Research Effort - We believe that there is a reasonable expectation that products useful to both the NRC and the industry will be developed if the present program is completed. We further believe that completion of this program meets the benefit/cost test when compared with the expected benefits of many other research activities that have been, and are continuing to be, supported by the staff.

We see a strong analogy between the present status of organizational factors research and the status of PRA methodology 20 years ago when the Reactor Safety Study, WASH-1400, was begun under the leadership of the AEC. There were many, both within the NRC and industry, who argued at the time that PRA was a nice theoretical exercise, but would never have practical uses. Today, PRA is employed as an extremely valuable, multi-use tool by both the NRC and the regulated industry. Without this initial leadership by the agency, it is doubtful that PRA would be at today's state of development.

We believe that it is likely that the organizational factors measurement tools that are currently under development and their possible integration into PRAs will play an important role in nuclear power plant safety technology in the years to come. We do expect that it will be necessary, just as it was with the development of PRA, for the NRC and industry to expend additional resources on organizational factors research.

There are considerable demands presently being placed on staff and licensee resources in such activities as the SALP Program and Diagnostic Team Inspections. For licensees, the periodic INPO evaluations create additional demands. If appropriately validated organizational factors measurement tools can be developed, it would be possible to optimize the use of staff and licensee resources in assessing licensee organizational performance. The present staff approach in assessing licensee organizational performance does not have an appropriately validated basis and is subject to legal challenge (such a challenge has already been made with respect to the SALP Program). Continuing this research program to provide validated organizational factors measurement tools has the potential of providing the staff with a much more defensible basis for its SALP Program and Diagnostic Team Inspections.

After organizational factors measurement tools become available, it will be possible to undertake completion of the next step; the

modeling of organizational factors into PRAs. If this modeling can be done in a credible manner, it would then be possible to assess how risk is apportioned between hardware and human performance. This would provide much needed insight into the manner in which NRC research efforts and inspection and enforcement resources should be allocated. It would also assist the staff and licensees in evaluating and correcting risk-significant weaknesses in their organizations.

We do not, however, believe that the integration of organizational factors into PRA should be the main focus of the present research program. Due to the complex, amorphous, and temporal nature of organizational performance, this objective may not be attainable. Rather, we believe that the emphasis should be on providing organizational effectiveness measurement tools to help the staff and the utilities better design and manage their organizations and to help the NRC make better judgments about the performance of licensee organizations. If the present integration efforts produce useful PRA input, so much the better. (We do believe that progress has been made by the researchers involved in this effort and recommend that this work be continued.)

Finally, we believe that the manner in which this research program has been carried out by the staff is representative of a serious generic problem that the staff has in dealing with complex issues that cut across staff organizational boundaries. We recommend that the EDO review the manner in which the various elements of the staff collaborated in developing the research objectives and in providing consistent guidance to the organizational factors research contractors. We expect such a review to lead to improved staff policy guidance on the coordination of future research efforts of this nature.

Reference:

SECY-93-020, dated February 1, 1993, for the Commissioners, from James M. Taylor, Executive Director for Operations, Subject: Review of Organizational Factors Research

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

March 18, 1993

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

Dear Chairman Selin:

SUBJECT: COMPUTERS IN NUCLEAR POWER PLANT OPERATIONS

During the 395th meeting of the Advisory Committee on Reactor Safeguards, March 11-12, 1993, we discussed the staff's progress in defining the regulatory requirements for digital instrumentation and control systems. During this meeting, we had the benefit of discussions with members of the NRC staff.

We have now had a long series of meetings, and have heard from many relevant people, but by no means all. To some extent our input has been biased in the direction of people, groups, and organizations who have experienced problems, and we have not heard from the legions of organizations who have successfully made the move into the computer world. It is important not to develop a tabloid mentality about new technology, i.e., aberrations from the norm treated as if they were the norm.

A first observation is that many of the anecdotes about catastrophic failures of major computer systems refer to systems far larger than those of interest here. Even the software systems on the C-17 aircraft, written in nearly a dozen languages for nearly a dozen machines, are far larger than any of relevance to the nuclear business. The Strategic Defense Initiative dispute is even less relevant. So we have to maintain perspective about scale.

A second observation is that computerization provides an opportunity, not a threat. The extraordinary reliability of electronic systems (unless abused), their potential for continuous and extensive self-testing in real time, their potential for relatively painless upgrades as experience accumulates, their ability to cover an enormous function space and to accommodate unseemly amounts of input data, their remarkable immunity to wear

(few, if any, moving parts)—all these provide the potential for safety enhancement. Much of our input from the staff has been devoted to the negative aspects of computerization, as if it were a disease to be kept in check.

A related observation is that the transition to computerized operation, control, instrumentation, support, recordkeeping, and maintenance procedures and records, is inevitable. The job of the NRC is not to manage or resist the transition, but to maintain a reasonable level of assurance that it is accomplished with proper accounting for the impact on safety. With any reasonable use of the technology the impact is expected to be large and positive.

The regulatory issues we have isolated in our series of subcommittee meetings fall broadly into two categories. One is a consequence of lack of nuclear regulatory experience with modern electronics, especially computers, leading to both extraordinary conservatism relative to unfamiliar accident sequences, and the application to a new technology of review methods and nomenclature derived from old habit and experience. The second is a collection of genuinely new problems associated both with the complexity of the new technology and with the consequent difficulty of assessing (as distinguished from assuring) its level of safety. We deal with these in order.

Failures of computerized systems (excluding fans, hard disks, and other mechanical components) do not follow the traditional bathtub curve of infant mortality, stable performance, and then wearout. Electrons don't wear out. Both in electronic hardware and software there tends to be a period of infant and young adult mortality (to which we will return), with performance and reliability gradually improving with time simply through natural selection—bugs are ironed out through experience and through extensive testing. There is no later period of wear, so there is no place for the regulatory and maintenance procedures associated with that part of the reliability pattern. Further, self-testing can provide constant assurance of full functionality of the electronics.

As a consequence, however, there has been little progress in applying the methods of probabilistic risk analysis, on which we have become so heavily dependent for mechanical, hydraulic, and electromechanical systems, to computer systems. Indeed the semiconductor components of the computerized systems are inherently so reliable that high-temperature life-testing is the only means available, in most cases, for generating any failures at all. Whereas one can generate probabilities for the existence of perinatal defects, there is no such thing as a probability per unit

time for the development of disease. Nor does in-service inspection play the same role.

These are important points, because the concepts of reliability and reproducibility differ, and the testing and verification procedures used depend on which is to be assured. A mechanical component with a presumed reliability of 10^{-3} failures per demand can be tested a few thousand times to assure that level of reliability, but a software-based system with a hidden bug that will be revealed in the event of an unlikely input configuration can be tested without failure until the cows come home, but will still always fail with that particular input. Interest has therefore to be directed at the probability that there is such a hidden bug, and the probability that some other circumstance may generate the unfortunate input. Neither of these probabilities will be discovered by repetitive testing under normal conditions. Randomized input testing can tell one something about the former probability, but not the latter. It is therefore misleading to bandy failure probabilities around, as if they had the same meaning as they do for familiar mechanical and electrical components. It also makes the direct comparison of computerized system reliability with the reliability of older technology more difficult.

These and other considerations mandate a format adjustment for the regulatory system, and such changes tend to be painful. What we have seen here is an unfortunate effort to cling to the old ways, to the point of asking that all digital systems have analog backups—not because the latter are better or more reliable, but because they are more familiar to the regulator and therefore easier to regulate. That alone could place an unwarranted burden on those seeking to improve safety by updating technology.

The second category of issues follows from the undoubted fact that computerized systems do indeed introduce unfamiliar failure modes, which require both recognition and palliative measures. Too much attention appears to have been concentrated on a microcosm of the more recognizable of these matters, specifically vulnerability of digital systems to electromagnetic interference (a subject on which there is enormous military expertise, largely untapped by the NRC staff), and the fact that replicated defective software (like replicated defective hardware) can be the source of common-mode failures. Both of these are real issues, but, in our judgment, not the central ones.

Let us first consider software issues. The literature is full of examples of cases in which carefully written and tested software still contains errors. Indeed it is doubtless true, though in

principle unprovable, that any large program that has not undergone a formal verification and validation (V&V) contains yet undiscovered errors. Lest there be confusion, it is well to be quantitative about the problem of implementing a function in software.

The simplest of all digital programs might generate a logic function, a mapping that accepts a number of binary inputs (say n) and generates a single binary output—a signal that might, in turn, activate a pump or a valve or some other sequence of events. Such a logic function has 2^n possible input states, over a thousand for $n=10$ and over a million for $n=20$. These are not unreasonable numbers of input states, because the input of a single number to one percent accuracy requires seven (usually more) binary inputs. Since each input state can have either output state (on/off), that means that even a modest eight-input binary converter of this sort can represent 2^{256} or 10^{77} different logic functions. A defect (either hardware or software) can change the desired function into any of the others. It is therefore reasonable to expect to test the system to make sure that it performs as designed, but not reasonable to expect to explore, by brute force, all consequences of all possible defects. The point is only strengthened if one has more complex outputs than just a single bit.

If, therefore, the requirements specified for the system describe the full mapping of the input space to the output space, special methods will be required to verify that this has been accomplished correctly. Such methods exist, and are applicable to relatively simple software packages. When formal V&V is possible, it provides assurance that the code, as written, correctly implements the formal specifications laid upon the design. When it is not possible (because the code is too long or too complex), there are many alternatives, but none of them provides the kind of assurance of code fidelity that is provided by formal V&V.

There appears to be a consensus among the experts we have consulted that the safety-related software in nuclear power plants is within reach of formal V&V methods, and that the potential for serious error lies more in incorrect expression of the specifications than in incorrect programming. Formal V&V can assure that the code correctly expresses the specifications, but not that the specifications are correct. In either case, it would appear that the staff emphasis on the possibility of common-mode errors in code segments used in different parts of the instrumentation and control system is misdirected. We continue to see an urgent need for staff augmentation with people experienced in thinking in the terms outlined above.

We believe that the experience of other industries that have accepted the progress has been characterized, almost without exception, by increases in efficiency and reliability, and by concomitant decreases in cost. (While the latter is not the NRC's business, it remains true that resources and attention released from unproductive safety concerns may, at least in part, find their way to better use.) There are genuine safety issues in this transition, of which one unfamiliar one is surely the requirement, in order to generate verifiable software, for precise no-nonsense attention to the specification of the functions to be implemented by the software.

The gist of our concerns is that the regulatory procedures developed during the decades preceding the full flowering of the electronic revolution (which may not yet have occurred) are inappropriate to the regulation of computerized functions in nuclear power plants. (This is true for both hardware and software--too much emphasis on the distinction is not helpful.) As a consequence, the staff has been dealing with the problems that have shown up so far on an ad hoc basis, applying methods created for each problem, with little underlying methodology. That has resulted in such distractions as the analog-to-digital conversion problem, the overemphasis on electromagnetic interference problems, the singling out of software common-mode failure as a central issue, etc., all without a framework into which the broad issues of regulatory emphasis and consistency can be fitted. We can cavil about the specific staff approaches to each of these, but the central issue is that neither the staff nor the Commission has established what could be described as a standard review plan or even a regulatory guide that could help both the staff and the industry know what is expected of them. A statement of the applicable standards ought to precede, not follow, their application. Without such a definition of objectives, coherence is an inevitable victim.

What, then, do we recommend? We frankly doubt that a coherent and effective review plan for computerized applications in nuclear power plants will be produced by the staff, the Commission (whose job is at a higher policy level), or the Committee (which is limited in both resources and expertise). Still, if one believes (as we do) that it needs to be done, it will be necessary to bring in outside help. It was in that context that we initiated our long series of subcommittee meetings on the subject. Our recommendation is that a workshop and study (with a charter to produce such a plan) be commissioned to be done by the National Academies of Sciences and Engineering. To derive maximum benefit from such a

study, there should be appropriate participation by key senior members of the staff.

Additional comments by ACRS Members James C. Carroll and Carlyle Michelson are presented below.

Sincerely,

Paul Shewmon
Chairman

Additional Comments by ACRS Members James C. Carroll and Carlyle Michelson

We agree with most of the technical observations made in this report. However, we disagree with the report's recommendation that a workshop and study be undertaken by the National Academies of Sciences and Engineering for the purpose of developing a review plan for computerized applications in nuclear power plants. Contrary to the view of our colleagues, we believe that the staff and its consultants are making satisfactory progress toward developing a "coherent and effective" review plan. Ideally, such a plan should have been developed in advance of the receipt of applications for the use of this rapidly changing technology. As a practical matter, it has been necessary for the staff to interact with the first group of applicants proposing computerized systems in order to gain an understanding of these systems. This has been a necessary first step before a generic review plan can be developed. Our view is that the proposed National Academies of Sciences and Engineering workshop and study would add little to the process of developing a staff review plan at this point in time.

We note that the staff has attended the series of ACRS subcommittee meetings on computerized applications in nuclear power plants that form the basis for this Committee report. In addition, the staff is planning to sponsor a workshop this fall and plans to obtain ACRS feedback on speakers and topics to be covered.

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

June 18, 1993

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

Dear Mr. Taylor:

During the NRC staff's May 14, 1993, meeting with the Commission, Mr. Russell, NRR, stated that material relevant to a number of international meetings on digital instrumentation and control matters had been provided to ACRS. Despite a diligent search, we have been unable to locate any such material (except for documents associated with the Japanese meeting in which an ACRS member participated, and a trip report on the Enlarged Halden Program Group Meeting from RES). We would appreciate receiving copies of all this material, so we can better discharge our advisory responsibilities to the Commission. This should include the information (trip reports, minutes, attendees, etc.) documenting the proceedings, participants, agenda, and lessons learned from these meetings.

Sincerely,

A large, stylized handwritten signature of J. Ernest Wilkins, Jr. is written over the typed name and title.

J. Ernest Wilkins, Jr.
Chairman

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

November 16, 1993

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

Dear Chairman Selin:

SUBJECT: COMPUTERS IN NUCLEAR POWER PLANT OPERATIONS

On March 18, 1993, we wrote you a report on the NRC staff approach to regulation of computers in nuclear power plant operations and upgrades. While there were many specific observations and suggestions in that letter, it ended by concluding that a fresh start was called for in developing an effective approach to this new and difficult subject, and recommended that you ask the National Academies of Sciences and Engineering to conduct a workshop directed at this end.

In the interim the staff has conducted its own workshop on digital systems, with the help of the National Institute of Standards and Technology, on September 14-15, 1993. Some of us attended that workshop, and our Chairman gave introductory remarks. It is therefore appropriate to ask whether that workshop served as a reasonable substitute for our earlier recommendation. We have concluded that it did not.

To begin, it was not a workshop, in the usual sense of the word. It was organized much as a technical session of a learned society, with a variety of relatively disconnected speeches by experts, limited opportunity for questions from the audience, and only a little opportunity for the experts to discuss the issues with each other.

The recommendation in our earlier letter was based on the belief that an open-minded approach, using the wealth of expertise in the outside world, might help to supply the framework on which a coherent regulatory structure might be hung. Wrangling over specific details of the staff position, like the requirement for hard-wired redundancy, or concentration on electromagnetic interference, could lead to a compromise animal, half fish and half cat, with little underlying rationale.

Based on our observation of the staff workshop, and discussions with our foreign colleagues during the recent Quadripartite Meeting of Advisory Committees, we have concluded that our recommendation to seek help outside, with a different format, remains appropriate.

The Honorable Ivan Selin

2

November 16, 1993

The NRC can muddle through the next few years on current momentum, but lack of an underlying rationale will ultimately exact a price, perhaps a high one. There are deep issues of regulatory philosophy here, and a case-by-case approach will continue to ignore them.

We repeat our original recommendation.

Sincerely,

A handwritten signature in cursive script that reads "J. Ernest Wilkins, Jr." The signature is written in dark ink and is positioned above the printed name.

J. Ernest Wilkins, Jr.
Chairman

Reference:

Report dated March 18, 1993, from Paul Shewmon, ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: Computers in Nuclear Power Plant Operations

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

December 20, 1993

MEMORANDUM TO: L. Joseph Callan
Acting Associate Director for Projects
Office of Nuclear Reactor Regulation

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

SUBJECT: PROPOSED SUPPLEMENT 6 TO GENERIC LETTER 89-10,
"INFORMATION ON SCOPE, GROUPING, PRIORITIZATION,
SCHEDULE, AND PUBLIC QUESTIONS"

During the 404th meeting of the Advisory Committee on Reactor Safeguards, December 9-11, 1993, the Committee decided not to review the proposed Supplement 6 to Generic Letter 89-10. The Committee has no objection to the staff proposal for issuing Supplement 6 to Generic Letter 89-10. The Committee appreciates your efforts in keeping it apprised regarding this matter.

A handwritten signature of John T. Larkins in cursive script.

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

Reference:

Memorandum dated June 2, 1993, from G. Marcus, NRC, for J. Larkins, ACRS, Subject: Forwarding of Proposed NRC Supplement 6 to Generic Letter 89-10, "Information on Scope, Grouping, Prioritization, Schedule, and Public Questions"

cc: S. Chilk, SECY	A. Gody, Jr., NRR
J. Taylor, EDO	R. Kiessel, NRR
J. Blaha, EDO	T. Scarbrough, NRR
J. Sniezek, EDO	H. Pastis, NRR
M. Taylor, EDO	E. Beckjord, RES
T. Murley, NRR	C. Heltemes, RES
J. Wiggins, NRR	G. Sege, RES
G. Marcus, NRR

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

May 20, 1993

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

Dear Mr. Taylor:

SUBJECT: DRAFT REPORT OF THE PRA WORKING GROUP

During the 397th meeting of the Advisory Committee on Reactor Safeguards, May 13-15, 1993, we discussed the draft report prepared by the NRC staff's PRA Working Group to document its activities during the past year and to provide recommendations on the use of PRA by the staff. Our Subcommittee on Probabilistic Risk Assessment held a meeting on May 11, 1993, to discuss this matter. During these meetings, we had the benefit of discussions with representatives of the staff.

The PRA Working Group provided us with a copy of the April 1993 draft of its proposed report to you, and the comments of its External Review Group on an earlier draft that we have not seen.

We believe that the PRA Working Group is carrying out its mission within the defined scope of the Program Plan. Also, much of the Working Group's effort and material will be useful. However, judging from the questions raised during our briefings, we think that the report is currently inadequate. It has taken nearly two years to reach this point, and the investment of further and expanded effort could be cost-effective. In retrospect, it may well be that the problems derive from the limited scope of the Program Plan.

The PRA Working Group was made aware of our concerns during the meetings. We would like, therefore, to defer our detailed recommendations until another round of external review has occurred. If the effect is to be substantial and positive, the report should have a minimum of rough edges.

The PRA Working Group is relatively small, composed of part-time players, few of whom have any direct and substantial experience participating in a real PRA. The group of four external reviewers, though uniformly of high quality, includes only one with substantial direct PRA experience. And we have seen little evidence that the vast industry experience in living with PRA has been tapped in any substantial way by the PRA Working Group.

May 20, 1993

We are further concerned that resources are being expended on efforts to define what PRA is, rather than on how it should be used within the agency. Substantial literature exists which defines what PRA is, and identifies the availability of appropriate data and models.

One of the external reviewers said of the earlier draft, "The fundamental questions of why and how NRC should use PRA have not been answered." The PRA Working Group considers these strategic questions outside the scope of the Program Plan. Yet, without attention to that overriding issue, this effort remains misdirected.

We believe that this issue can only be addressed by a collaboration between the PRA Working Group and high-level NRC management. The issue is of sufficiently high policy significance that it should not be delegated to middle management.

Sincerely,

A handwritten signature in cursive script that reads "Paul Shewmon".

Paul Shewmon
Chairman

Reference:

Memorandum dated April 22, 1993, from Eric S. Beckjord, Director, Office of Nuclear Regulatory Research, for James M. Taylor, EDO, Subject: Draft Report of the PRA Working Group (Includes External Reviewers' Comments on Previous Draft)

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

November 10, 1993

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

Dear Chairman Selin:

SUBJECT: DRAFT FINAL REPORT OF THE PRA WORKING GROUP

During the 403rd meeting of the Advisory Committee on Reactor Safeguards, November 4-6, 1993, we heard presentations by the NRC staff on the draft final report of the PRA Working Group and its recommendations to the Commission. We also had the benefit of the documents referenced, of which we call special attention to the November 2, 1993 letter of the NRC Office Directors to the Executive Director for Operations.

In general, we were favorably impressed by the report, and of course gratified that the final version took account of many of the concerns expressed by the external reviewers and by us. In some cases, the responses were aspirational (i.e., to the pivotal concern that there is as yet no NRC policy on how PRA should be used in regulation, the report acknowledges that that is important and needs to be addressed), but even aspirational responses are better than denials that there is a problem. What really matters, of course, is the extent to which NRC will in fact enhance its capabilities, tune its regulatory activities to the risk posed by the objects of regulation, and adjust its life style to the new awareness of the implications of probabilistic analysis.

In this context we welcome the November 2, 1993 letter mentioned above, which records the intent of the Office Directors to develop a plan for the application of PRA throughout the agency, and to do so by December 30, 1993. In such a short time span, especially at this time of year, it is not possible to do more than establish a program plan, and make the commitment of resources. Given the magnitude of the job, the history of inconsistency and unevenness in the use of PRA, the frequent misunderstandings, etc., those resources will have to be substantial if the job is to be taken seriously. We have to reserve judgment until we can see if the actions match the words.

Still, we think that the PRA Working Group has done a creditable job, especially given the limited resources it had available, and we are heartened by the positive response accorded its report by the senior staff.

Some of the problems left for the future are, though acknowledged, extremely difficult and fundamental. A central issue since the beginning is to find a mechanism for the incorporation of risk-based, and therefore probabilistic, considerations into a deterministic regulatory structure. The Committee has only hinted at the existence of techniques for doing this, and the question is left entirely open by the PRA Working Group. It will not be simple, especially in an agency whose staff has limited training and experience in such matters.

We are therefore pleased that the Working Group has produced a valuable report, and that the senior staff appears to be taking it seriously. After the battle at El Alamein in World War II, Winston Churchill said that it was not the end, nor even the beginning of the end, but that perhaps it was the end of the beginning. We have the same cautious hope. We remain interested in this activity, and would like to be kept aware of the progress.

Sincerely,

J. Ernest Wilkins, Jr.
Chairman

References:

1. Memorandum dated October 8, 1993, from Warren Minners, NRC, for John T. Larkins, ACRS, Subject: PRA Working Group Draft Final Report (Draft Predecisional)
2. Memorandum dated November 2, 1993, from NRC Office Directors (NRR, RES, AEOD, NMSS) for James M. Taylor, NRC Executive Director for Operations, Subject: Agency Directions for Current and Future Uses of Probabilistic Risk Assessment (PRA)
3. Letter dated May 20, 1993, from Paul Shewmon, ACRS Chairman, to James M. Taylor, NRC Executive Director for Operations, Subject: Draft Report of the PRA Working Group
4. Letter dated July 19, 1991, from David A. Ward, ACRS Chairman, to the Honorable Ivan Selin, NRC Chairman, Subject: The Consistent Use of Probabilistic Risk Assessment

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

April 30, 1993

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

Dear Chairman Selin:

SUBJECT: STAFF INITIATIVES TO REVISE THE SYSTEMATIC ASSESSMENT OF
LICENSEE PERFORMANCE PROGRAM

During the 396th meeting of the Advisory Committee on Reactor Safeguards, April 15-17, 1993, we discussed with representatives of the NRC staff and NUMARC the staff's final recommendations for changes to the Systematic Assessment of Licensee Performance (SALP) Program, as delineated in SECY-93-090. We also had the benefit of the documents referenced.

Since SECY-93-090 was a predecisional document before the Commission's April 15, 1993 SALP briefing, the NUMARC representatives did not have an opportunity to review it before our discussion.

In a number of our past reports, we have provided comments and recommendations to the Commission based on our assessment of the SALP Program. In general, we have agreed with the longstanding industry position that major changes were needed to correct serious problems with the Program. A major thrust of our past comments and recommendations was that the staff inappropriately uses the Program as a means of imposing its demands and expectations (beyond what is required by the NRC's basic regulatory requirements) on nuclear power plant licensees. We have argued for a more effective set of checks and balances on the SALP Program and more NRC senior staff management involvement in monitoring its implementation.

The staff has evaluated comments related to the SALP Program that it received during its 1989 Regulatory Impact Survey and, in response to a Staff Requirements Memorandum, it developed "preliminary conclusions for changes to the SALP program" as described in SECY-92-290. The staff then sought public comment on these proposed changes. Additional changes are now being proposed by the staff as described in SECY-93-090.

We have the following comments and recommendations on this SECY paper:

- We agree with the staff that an effective, integrated program for periodically assessing licensee performance is a necessary regulatory tool.
- We believe that the changes to the SALP Program that the staff is proposing will prove to be beneficial. However, we continue to point out that many of the important changes are aspirational in nature. Good intentions do not always result in improved and more effective regulation. Accordingly, we recommend that the Commission establish a periodic feedback mechanism so that it can monitor the anticipated staff progress in improving the SALP Program. One such mechanism would be to conduct another Regulatory Impact Survey in one to two years after these changes to the SALP Program have been implemented.
- We recommend that the Commission formalize an appeal process that would permit a licensee to bring grievances regarding the application of the SALP Program to the attention of senior staff management without fear of retribution.
- We are persuaded by the staff's arguments that the objectives of the SALP Program require the use of a numerical grading system for the consolidated SALP Functional Areas.

We expect to interact with the staff and the industry on this important matter as experience is gained with the SALP Program.

Additional comments by ACRS Members James C. Carroll, Harold W. Lewis, and Charles W. Wylie are presented below.

Sincerely,

Paul Shewmon
Chairman

Additional Comments by ACRS Members James C. Carroll, Harold W. Lewis, and Charles W. Wylie

We are in agreement with the Committee's report with the exception of the comment that the "objectives of the SALP Program require the use of a numerical grading system for the consolidated SALP Functional Areas." We believe that many of the internal and external difficulties with the Program would be lessened if the grading system were eliminated.

We note that INPO's periodic evaluation program does not use a numerical grading system for individual plant functional areas.

Their program appears to be effective in communicating the results of the evaluations to the utilities. We also note that the staff's proposal is inconsistent in that the Plant Support Functional Area now comprises several important rating categories (including some that were previously classified as individual Functional Areas). Use of a single grade for the Plant Support Functional Area does not provide the numerical grades for these important categories that the staff claims it needs "in its allocation of resources to oversee, inspect, and assess licensee performance."

We recommend that the staff develop a pilot program (perhaps centered in one region) to test the effectiveness of the Program without the use of a numerical grading system. Recall that on December 21, 1989, the ACRS recommended that the Program be suspended, and that no new ratings be issued until it is fixed. Soon thereafter the Commission considered eliminating numerical ratings entirely, and the motion was defeated on a tie vote.

References:

1. SECY-93-090, dated April 6, 1993, for the Commissioners, from James M. Taylor, Executive Director for Operations, NRC, Subject: Systematic Assessment of Licensee Performance (SALP) Program
2. Letter, dated December 18, 1992, from Ivan Selin, NRC Chairman, to Joe F. Colvin, NUMARC, responding to NUMARC's October 20, 1992, letter on the SALP Program
3. Letter, dated October 20, 1992, from Joe F. Colvin, NUMARC, to Ivan Selin, NRC Chairman, providing industry views on the SALP Program
4. Letter, dated October 9, 1992, from Douglas S. Reynolds and David S. Repka, Winston & Strawn, to David L. Meyer, U.S. NRC, regarding SALP Program
5. SECY-92-290, dated August 19, 1992, for the Commissioners, from James M. Taylor, Executive Director for Operations, NRC, Subject: Systematic Assessment of Licensee Performance (SALP) Program
6. Memorandum dated December 20, 1991, from Samuel J. Chilk, Secretary, for James M. Taylor, Executive Director for Operations, NRC, Subject: SECY-91-172 - Regulatory Impact Survey Report - Final
7. Report from David A. Ward, ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: The Staff's Recommendations on the Regulatory Impact Survey Report, September 10, 1991
8. Letter, dated December 11, 1990, from Zack T. Pate, Institute of Nuclear Power Operations, to Kenneth M. Carr, NRC Chairman, providing comments on the results of the NRC Regulatory Impact Survey and ACRS comments on regulatory coherence
9. Report from Carlyle Michelson, ACRS Chairman, to Kenneth M. Carr, NRC Chairman, Subject: Reevaluation of the SALP Program, September 12, 1990

10. Letter, dated September 4, 1990, from Joe F. Colvin, NUMARC, to Harold W. Lewis, ACRS, providing comments on proposed changes to the SALP Program
11. Report from Carlyle Michelson, Acting Chairman of ACRS, to Kenneth M. Carr, NRC Chairman, Subject: Coherence in the Regulatory Process, December 21, 1989

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

April 23, 1993

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

Dear Mr. Taylor:

SUBJECT: PROPOSED FINAL VERSIONS OF REGULATORY GUIDES FOR
IMPLEMENTING REVISED 10 CFR PART 20, "STANDARDS FOR
PROTECTION AGAINST RADIATION"

During the 396th meeting of the Advisory Committee on Reactor Safeguards, April 15-17, 1993, we discussed the proposed final versions of the three referenced regulatory guides that provide guidance for implementing some of the requirements of the revised 10 CFR Part 20. Our Subcommittee on Occupational and Environmental Protection Systems and a Working Group of the Advisory Committee on Nuclear Waste also discussed these guides with representatives of the Office of Nuclear Regulatory Research (RES) during a joint meeting on March 26, 1993. ACRS and ACNW had provided comments on the earlier versions of these guides in letters dated October 17, and October 23, 1991.

We believe that these guides provide an effective implementation strategy and should prove very useful to the licensees and regulatory authorities. These guides reflect careful consideration by the RES staff of both our earlier comments and the public comments. We concur in the regulatory positions of these guides and recommend that they be issued expeditiously.

Sincerely,

Paul Shewmon

Paul Shewmon
Chairman

References:

1. Regulatory Guide 8.N.10, "Control of Access to High and Very High Radiation Areas in Nuclear Power Plants," August 1992
2. Regulatory Guide 8.9, Revision 1, "Acceptable Concepts, Models, Equations and Assumptions for a Bioassay Program," March 1993
3. Regulatory Guide 8.37, "ALARA Radiation Programs for Effluents From Materials Facilities," March 1993
4. 10 CFR 20, "Standards for Protection Against Radiation," revised on May 21, 1991

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

July 15, 1993

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

Dear Mr. Taylor:

SUBJECT: PROPOSED DRAFT REGULATORY GUIDES, DG-1023, "EVALUATION OF REACTOR PRESSURE VESSELS WITH CHARPY UPPER-SHELF ENERGY LESS THAN 50 FT-LB," AND DG-1025, "CALCULATIONAL AND DOSIMETRY METHODS FOR DETERMINING PRESSURE VESSEL NEUTRON FLUENCE"

During the 399th meeting of the Advisory Committee on Reactor Safeguards, July 8-9, 1993, we discussed the subject draft regulatory guides. Our Subcommittee on Materials and Metallurgy examined these guides in detail at a meeting on June 29, 1993. During these meetings, we had the benefit of discussions with representatives of the NRC staff. We also had the benefit of the document referenced.

The need for these proposed guides was highlighted during the evaluation of the integrity of the Yankee Rowe reactor pressure vessel. We believe that these guides should prove useful to the licensees and regulatory authorities, and recommend that they be issued for public comment. We would like an opportunity to review the proposed final version of these guides after the public comments have been reconciled and before they are published in final form.

Sincerely,

A handwritten signature in black ink, reading "J. Ernest Wilkins, Jr.", is positioned above the typed name.

J. Ernest Wilkins, Jr.
Chairman

Reference:

Memorandum dated June 10, 1993, from Lawrence C. Shao, Office of Nuclear Regulatory Research, for John T. Larkins, ACRS, Subject:

Mr. James M. Taylor

2

July 15, 1993

Request for ACRS Review of Proposed Draft Regulatory Guides, with enclosures:

- DG-1023, "Evaluation of Reactor Pressure Vessels with Charpy Upper-Shelf Energy Less than 50 ft-lb," and
- DG-1025, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence"

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

June 18, 1993

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

Dear Chairman Selin:

SUBJECT: POLICY STATEMENT ON TECHNICAL SPECIFICATIONS IMPROVEMENTS
FOR NUCLEAR POWER PLANTS

During the 398th meeting of the Advisory Committee on Reactor Safeguards, June 10-11, 1993, we reviewed the NRC staff's draft Final Policy Statement on Technical Specifications Improvements for Nuclear Power Plants, as originally presented in SECY-93-067. We also reviewed a revised draft of this Policy Statement which is responsive to the Commission's comments included in the Staff Requirements Memorandum dated May 25, 1993. During our meeting, we had the benefit of discussions with representatives of the NRC staff. We also had the benefit of the documents referenced.

In SECY-93-067, the staff recommended that the draft Final Policy Statement be published for a 90-day public comment period. However, the Commission approved publication of the Policy Statement in final form, subject to the following comments:

1. The Commission directed the staff to prepare a rulemaking package that would codify the four criteria contained in SECY-93-067, delineating those aspects of nuclear power plant design and operation that should be included in Technical Specifications. (We note that the staff has proposed the use of these same criteria for establishing plant systems and components requiring an "effective program" under the license renewal rule.) The Commission also directed the staff, in developing the proposed rule, to ensure that the voluntary nature of the improved Standard Technical Specification program be preserved and that the Federal Register notice indicate that public comments on the proposed rule will be welcomed, considered, and addressed during preparation of the final rule. The staff was also directed to prepare any regulatory guides needed to implement this rule.

We agree with the above actions by the Commission and believe that the staff has appropriately modified the Policy Statement in response to the Commission's comments. The staff, of

June 18, 1993

course, needs to proceed with the other matters covered by these comments.

2. The Commission also directed the staff to modify the Policy Statement to clarify how it intends to utilize probabilistic risk assessments (PRAs) in its review of Technical Specification change requests involving Criterion 4 - "A structure, system, or component which operating experience or probabilistic safety assessment [PRA] has shown to be significant to public health and safety." The Commission apparently has no problem with this criterion, but believes that if the results of a PRA indicate that Technical Specifications can be relaxed or removed, a deterministic review should be performed. If the results of the deterministic review also support relaxing or removing the Technical Specifications, the staff should not preclude such action.

We agree with the view expressed by the Commission on this issue. The staff believes that it has responded to the Commission's comment in the modified Policy Statement by clarifying how it intends to utilize PRA in its review of Technical Specification change requests. We believe that the staff needs to provide more detailed guidance on the definition of "significant to public health and safety." This additional guidance should probably appear in the implementing regulatory guide(s).

This problem with Criterion 4 also exists in a number of recent staff initiatives (obvious examples are structures, systems, and components to be covered by the Maintenance Rule and the staff's reluctance to define "vulnerabilities" with respect to the Individual Plant Examination program).

Many problems related to the use of PRA by the NRC staff were described in our May 20, 1993 letter concerning the "Draft Report of the PRA Working Group." The issue raised in the present report is in the same class.

Sincerely,

J. Ernest Wilkins, Jr.
Chairman

References:

1. SECY-93-067 dated March 17, 1993, for the Commissioners from James M. Taylor, Executive Director for Operations, NRC, Subject: Final Policy Statement on Technical Specifications Improvements

2 of 2

2. Memorandum dated June 3, 1993, from Brian K. Grimes, Office of Nuclear Reactor Regulation, for John T. Larkins, ACRS, Subject: Request for ACRS Review of Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors
3. Staff Requirements Memorandum dated May 25, 1993, from Samuel J. Chilk, Secretary, for James M. Taylor, Executive Director for Operations, NRC, Subject: SECY-93-067 - Final Policy Statement on Technical Specifications Improvements

UNITED STATES
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WASHINGTON, D. C. 20555

July 15, 1993

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

Dear Chairman Selin:

SUBJECT: REGULATORY REVIEW GROUP REPORT

During the 399th meeting of the Advisory Committee on Reactor Safeguards, July 8-9, 1993, we discussed the report prepared by the Regulatory Review Group (RRG) for public comment, and the proposed rulemaking to implement its recommendations on 10 CFR 50.54. Our Subcommittee on Regulatory Policies and Practices considered these matters during a meeting on July 7, 1993. During these meetings, we had the benefit of discussions with representatives of the NRC staff. We also had the benefit of the documents referenced.

Though the report was not entirely responsive to the charter of the RRG, we find that understandable, in view of both the grand scope of the project, and the limited time allotted for the job. We think the RRG has done well.

It is important to maintain some perspective, lest the central issues get lost in a sea of minutiae. We therefore will concentrate here on the broader policy issues.

It is almost axiomatic that regulation without a defined objective tends to be uncontrollable. This presumably led the Commission to promulgate its safety goals and quantitative health objectives in 1986. Unfortunately, in part because that Policy Statement paid no attention to the inevitable uncertainties that arise in the implementation of any quantitative policy, the goals provide little guidance to the staff in discharging its everyday responsibilities. We have indeed urged that their principal use be in judging the effectiveness of the set of deterministic regulations that serve as enforceable surrogates for the goals themselves. Confusing the issue is the question of "adequate protection," words that appear in a minor clause in the Atomic Energy Act, but which play a legal role in the implementation of the Backfit Rule. Continuing along the line from the fundamental (the safety goals), through the regulations (the measures intended to achieve the objective), one

finds at the next level of regulation a potpourri of commitments, understandings, and declarations intended to supplement the rules and regulations in assuring nuclear safety. It is to this level of regulation that the most important recommendation in the report is addressed. (There is of course another level, occasional informal direction of licensees by NRC staff, which is the subject of neither this letter nor the report it reviews.)

Over the years, there has developed, rightly or wrongly, the sense that simple enforcement of the rules and regulations is inadequate to assure satisfactory protection of the health and safety of the public, and there has accumulated a long list of both plant-specific and generic commitments, to which the licensees are bound. Indeed, in the debates about license renewal, one of the stumbling blocks has been the lack of a suitable definition of the current licensing basis on which renewal decisions will be based. Further, and this is the point addressed by the RRG, a licensee seeking relief from a commitment that goes beyond the rules and regulations must prepare a case for the action, and secure NRC permission for the change. The RRG proposes to change this procedure in two related ways, one declaratory, and one procedural, but each with such substantial implications that the changes can not be expected to go down easily.

The RRG proposes that the Commission declare that adherence to the rules and regulations that have evolved constitutes the fundamental condition laid upon a licensee under 10 CFR 50.54, and that the body of further commitments should be viewed as means to that end. This would then have the consequence that a licensee would have the right, while still fulfilling its fundamental obligation, to alter or change commitments that it deems unnecessary to meet the rules and regulations, without seeking prior NRC approval. NRC would of course have to be notified, and the rationale available. NRC could then object on the basis that the action may have brought the licensee into conflict with a rule or regulation, but only on that basis. Then the burden of proof would lie with the NRC to make its case. In this way, conformance to the rules and regulations would be the governing obligation of the licensee.

This would constitute a fundamental change, and is likely to receive a rather thorough set of reviews and analyses before it takes effect, so we think it premature to comment about the more detailed implementation recommendations contained in the report. They will surely change under scrutiny.

We think that the RRG recommendation is a substantial positive step, worth serious consideration by the Commission. We do not recommend that the RRG be continued past its scheduled dissolution,

July 15, 1993

but are concerned that natural resistance to change will bury one of the few recent proposals for substantial change, without due process. We therefore recommend that you take the steps necessary to move the recommendations to the next phase, which is more detailed consideration about how such a fundamental change might be implemented. In view of our earlier discussion of the relationships among the formal and informal elements of the regulatory structure, this would be a step toward coherence.

Sincerely,

A handwritten signature in cursive script, reading "J. Ernest Wilkins, Jr.", written in dark ink.

J. Ernest Wilkins, Jr.
Chairman

References:

1. U.S. Nuclear Regulatory Commission, Regulatory Review Group Report, Volumes 1-4, issued for public comment on May 28, 1993
2. Memorandum dated June 25, 1993, from Frank P. Gillespie, Regulatory Review Group, for John T. Larkins, Executive Director, ACRS, Subject: Proposed Rulemaking to Implement the Regulatory Review Group Recommendations on 10 CFR 50.54 (Draft Predecisional)

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

November 10, 1993

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

Dear Chairman Selin:

SUBJECT: DRAFT COMMISSION PAPER, "POLICY AND TECHNICAL ISSUES
ASSOCIATED WITH THE REGULATORY TREATMENT OF NON-SAFETY
SYSTEMS IN PASSIVE PLANT DESIGNS"

During the 403rd meeting of the Advisory Committee on Reactor Safeguards, November 4-6, 1993, we reviewed the NRC staff's positions and recommendations in the subject draft Commission paper, which reflects changes resulting from public comments on an earlier draft. We reviewed this earlier draft during our 400th meeting, August 5-6, 1993. Also, our Subcommittee on Improved Light Water Reactors reviewed this matter during a meeting on August 4, 1993. During this review, we had the benefit of discussions with representatives of the NRC staff and EPRI. We also had the benefit of the documents referenced.

The basic issue under review is that passive plant designs rely on passive safety systems to meet the regulatory requirements, but also include active non-safety systems as a first line of defense to reduce challenges to the passive safety systems in the event of transients or plant upsets. As this represents a departure from the current licensing approach, the draft Commission paper is intended to develop regulatory and review guidance for the AP600 and SBWR certification submittals.

In the draft Commission paper, the staff identified eight issues that pertain to the regulatory treatment of non-safety systems (RTNSS) for passive LWRs. We are in general agreement with the staff's positions and recommendations for resolving these issues, but have the following specific comments on three particular issues.

A. Regulatory Treatment of Non-Safety Systems

This specific issue has the same name as the general subject because it addresses an overall process for resolving the various issues. The overall process proposed by the staff would make innovative use of PRA to determine the risk significance of active non-safety systems with respect to meeting the ancillary safety goal on core-melt frequency, and a large release goal not fully defined. Reliability/availability "missions" for the active non-safety systems would be developed and regulatory oversight procedures applied that would depend on the assessed risk significance.

In general, we think the proposed RTNSS process is a bold and positive step in the direction of risk-based regulation. We recommend that the Commission approve this general process, and we encourage the staff to proceed with further development, to address some of our specific concerns, and to begin the implementation of the process. Our specific concerns are as follows:

1. The staff is still proposing the use of a "large release" frequency of 1×10^{-6} /yr as a "safety goal guideline." Since a different segment of the staff previously recommended abandoning this concept (we think for good reason), it is disturbing to see it being resurrected here. We believe the RTNSS process would be better served by use of a conditional containment failure guideline.
2. We believe that the risk significance of the active systems (as developed from the baseline and focused PRA) will be sensitive to the reliability values assumed in the PRAs for the passive systems. We are concerned that there does not exist a sufficient data base to establish appropriate reliability values for use in the proposed process.
3. We were told that the reliability/availability "missions" for the risk-significant active non-safety systems will, in fact, be reliability values. The proposed process is vague about how the review and regulatory audit processes can determine whether or not such reliability "missions" will have been met in the design and maintained during operation. We believe that the proposed review and audit processes, reliability assurance program, and implementation of the Maintenance Rule will not provide assurance that such "missions" have been met.

4. The document calls for generating uncertainty distributions for the PRA results. Since the only numerical goals mentioned were based on mean values, it is not clear to us how the uncertainties are to be used by the staff.

B. Definition of Passive Failure

The draft Commission paper identifies certain passive failures that could initiate accidents. Included are check valve failures, medium- or high-energy pipe failures, and valve stem or bonnet failures. We note that valve stem or bonnet failures are included as initiating failures for the passive plants. To the best of our knowledge, the staff does not postulate such failures as current licensing practice for evolutionary plants. If such a failure were postulated to occur in the outboard containment isolation valve for the reactor water cleanup system of the Advanced Boiling Water Reactor, and the postulated single active component failure results in a failure to close the inboard containment isolation valve, the final result would be an unisolated loss-of-coolant accident outside of the primary containment.

Concerning check valves, we support the staff position to redefine check valves (except for those whose proper function can be demonstrated and documented) in the passive safety systems as active components subject to the single failure consideration.

C. Reliability Assurance Program
(Issue E in the draft Commission Paper)

We are in substantial agreement with the staff proposal on the reliability assurance program (RAP). It is noted that this program represents a significant commitment of resources by the ALWR vendor and, even more, the COL applicant. The use of modern risk assessment methods in identifying the systems, structures, and components to be covered within this program, and hence the use of these resources, is an important feature of the staff approach. We continue to recommend that the RAP be integrated with implementation of the Maintenance Rule.

Sincerely,

J. Ernest Wilkins, Jr.
Chairman

References:

1. Draft Commission Paper (Undated), from James M. Taylor, NRC Executive Director for Operations, for The Commissioners, Subject: Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs, received July 21, 1993
2. Revised Draft Commission Paper (Undated), Subject: Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs, received November 4, 1993

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

December 20, 1993

MEMORANDUM TO: Warren Minners, Director
Division of Safety Issue Resolution
Office of Nuclear Regulatory Research

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

SUBJECT: DRAFT RULEMAKING PACKAGE ELIMINATING THE
EMERGENCY PLANNING ANNUAL EXERCISE

During the 404th meeting of the Advisory Committee on Reactor Safeguards, December 9-11, 1993, the Committee decided to postpone its review of the subject matter until after the public comments have been reconciled by the staff.

Reference:

Memorandum dated November 10, 1993, for J. Larkins, ACRS, from W. Minners, NRC, Subject: ACRS Review of Draft Rulemaking Package Eliminating the Emergency Planning Annual Exercise

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

cc: S. Chilk, SECY
J. Taylor, EDO
J. Blaha, EDO
M. Taylor, EDO
T. Murley, NRR
L. Callan, NRR
E. Beckjord, RES
C. Heltemes, RES
G. Sege, RES
M. Jamgochian, RES

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

April 23, 1993

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

Dear Chairman Selin:

SUBJECT: SECY-93-049, IMPLEMENTATION OF 10 CFR PART 54,
REQUIREMENTS FOR RENEWAL OF OPERATING LICENSES FOR
NUCLEAR POWER PLANTS

During the 395th and 396th meetings of the Advisory Committee on Reactor Safeguards, March 11-12 and April 15-17, 1993, we discussed with the NRC staff its proposal in SECY-93-049 for implementing the License Renewal Rule. During our April meeting, we also heard from representatives of NUMARC on this matter. We had the benefit of the documents referenced.

In a number of our past reports, we provided comments and recommendations on various aspects of the License Renewal Rule including our recommendation that this rule and the Maintenance Rule be better integrated in the interest of long-term coherence of the regulatory process. We have also commented that the operational phase reliability assurance program required of applicants licensed under 10 CFR Part 52 needs to be integrated with these two rules. Additionally, we have strongly opposed the staff's proposals to use license renewal as a means of dealing with such issues as electrical cable qualification and mechanical component fatigue life on the occasion of a plant's 40th birthday, when, in fact, these issues potentially affect presently operating plants that may or may not seek license renewal. We continue to support these views.

The staff's recent efforts to develop an approach to facilitate a more effective and efficient license renewal process that would rely heavily on use of the requirements of the Maintenance Rule appear to have the potential for making significant improvements in this process. However, we believe further dialogue between the staff and the industry on this matter is needed so that the many subtle issues involved in this approach are fully explored.

As we understand the present situation, the Commission's major concern is whether (1) the present License Renewal Rule can be legally construed to accommodate the staff's proposal in SECY-93-049 or (2) there is a need to revise or formally interpret the existing rule and its accompanying statements of consideration to

accurately reflect the course of action proposed by the staff. From a policy point of view, we believe that this Commission's thinking on the formulation of the ultimate implementation of the License Renewal Rule needs to be documented by a policy statement, interpretive rulemaking, or a revision to the rule and its statements of consideration. Based on our discussions with the staff and NUMARC, it appears that the needed time is available to revise the rule without significantly impacting licensees' schedules for making decisions regarding license renewal.

Sincerely,

Paul Shewmon
Chairman

References:

1. SECY-93-049, dated March 1, 1993, for the Commissioners, from James M. Taylor, Executive Director for Operations, NRC, Subject: Implementation of 10 CFR Part 54, Requirements for Renewal of Operating License for Nuclear Power Plants
2. Report from David A. Ward, ACRS Chairman, to James M. Taylor, Executive Director for Operations, NRC, Subject: Proposed Guidance for Implementation of the Maintenance Rule, 10 CFR 50.56, October 15, 1992
3. Report from David A. Ward, ACRS Chairman, to James M. Taylor, Executive Director for Operations, NRC, Subject: Proposed Regulatory Guide and Interim Standard Review Plan for License Renewal and a Related Branch Technical Position on Fatigue Evaluation Procedures, August 17, 1992

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

May 20, 1993

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

Dear Chairman Selin:

SUBJECT: BACKFIT RULE

During the 397th meeting of the Advisory Committee on Reactor Safeguards, May 13-15, 1993, we discussed the proposed options listed in SECY-93-086 on possible amendments to 10 CFR 50.109, to make its backfit provisions more flexible. During this meeting, we had the benefit of discussions with representatives of the NRC staff.

We are not persuaded that any action on the part of the Commission is warranted. The Commission has adequate authority to enforce actions that are appropriate in the public interest, and any weakening of the rule is likely to bring back the conditions the rule was originally designed to fix.

Sincerely,

A handwritten signature in cursive script that reads "Paul Shewmon".

Paul Shewmon
Chairman

Reference:

SECY-93-086, dated April 1, 1993, for the Commissioners, from James M. Taylor, Executive Director for Operations, NRC, Subject: Backfit Considerations

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

June 18, 1993

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

Dear Chairman Selin:

SUBJECT: SECY-93-113, ADDITIONAL IMPLEMENTATION INFORMATION FOR
10 CFR PART 54, "REQUIREMENTS FOR RENEWAL OF OPERATING
LICENSES FOR NUCLEAR POWER PLANTS"

During the 398th meeting of the Advisory Committee on Reactor Safeguards, June 10-11, 1993, we discussed with the staff its proposals in SECY-93-113 for clarifying the staff's approach described in SECY-93-049, Implementation of 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants." We had the benefit of the documents referenced.

More can be done to reduce the necessary scope of review of components subject to age-related degradation by giving full credit to the maintenance programs in place during the initial term of license. Where these maintenance programs have been determined by the NRC to be adequate in preserving safety during the original term of the license, they become part of the current licensing basis and, if continued, may be acceptable for managing age-related degradation during the license renewal period. If necessary, 10 CFR Part 54 should be revised to permit the staff to recognize these programs.

The staff has made substantial progress in clarifying how it will implement 10 CFR Part 54. It will now invite public comment. This should disclose whether the uncertainty that the industry representatives once believed to be present in the rule is now acceptably lowered. We will be interested in reviewing the public comments and the staff's resolution of these comments.

Sincerely,

J. Ernest Wilkins, Jr.
Chairman

References:

1. SECY-93-113 dated April 30, 1993, for the Commissioners from James M. Taylor, Executive Director for Operations, NRC, Subject: Additional Implementation Information for 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants"
2. SECY-93-049 dated March 1, 1993, for the Commissioners from James M. Taylor, Executive Director for Operations, NRC, Subject: Implementation of 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants"

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

April 26, 1993

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

Dear Mr. Taylor:

SUBJECT: IMPLEMENTATION GUIDANCE FOR THE MAINTENANCE RULE

During the 396th meeting of the Advisory Committee on Reactor Safeguards, April 15-17, 1993, we discussed with the NRC staff the status of its proposed implementation guidance for the Maintenance Rule, 10 CFR 50.65. We also heard from representatives of NUMARC on this matter and had the benefit of the documents referenced.

The staff's present plan is that this implementation guidance will be in the form of the Regulatory Guide entitled "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants" that endorses the NUMARC 93-01 document as an acceptable means of complying with the provisions of the Maintenance Rule. Both of these documents have been issued for public comment and the comments received have been analyzed by the staff. In addition, NUMARC conducted a validation and verification effort to test the guidance in the NUMARC 93-01 document by having a number of licensees apply it to their plants. The staff participated in this effort. We commend both the staff and NUMARC for their efforts in producing what appears to be a well-considered approach to implementation of the performance-based Maintenance Rule.

The process is now at a point where the staff and NUMARC are finalizing their respective documents with the expectation that they will be issued in final form by June 30, 1993. Contrary to what is stated in the draft of the regulatory guide, we do expect to review these documents when they are completed.

At this time, we have the following comments to offer:

- On many occasions, we have provided comments on the trigger-value approach proposed by the staff to resolve Generic Issue B-56, "Diesel Generator Reliability." The proposed regulatory guide for implementing the Maintenance Rule explicitly endorses the trigger value procedure for "monitoring emergency diesel generator (EDG) performance against EDG target reliability levels." It is categorically impossible to demonstrate the reliability of EDGs using this method. We remain strongly opposed to its use for this purpose and continue to recommend

that the staff's implementation guidance for the Station Blackout Rule, 10 CFR 50.63, be revised to deal with this issue. When this is done, the regulatory guide should be appropriately revised.

- We agree with the staff's approach in resolving our concerns regarding maintenance in power plant switchyards. We recommend, however, that appropriate plant management exercise control of all such switchyard activities to prevent the kind of unanticipated events that have occurred in the past.

Sincerely,

Paul Shewmon

Paul Shewmon
Chairman

References:

1. Draft Regulatory Guide DG-1020, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," November 1992
2. Draft of Final Regulatory Guide (formerly DG-1020), Regulatory Analysis and Backfit Analysis for 10 CFR 50.65, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants" (hand dated April 13, 1993)
3. Draft NUMARC 93-01, Revision 3, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," March 24, 1993

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

September 20, 1993

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

Dear Mr. Taylor:

SUBJECT: PROPOSED RULE AMENDING FRACTURE TOUGHNESS REQUIREMENTS FOR LIGHT WATER REACTOR PRESSURE VESSELS, PROPOSED RULE REGARDING REQUIREMENTS FOR THERMAL ANNEALING OF REACTOR PRESSURE VESSELS, AND DRAFT REGULATORY GUIDE ON FORMAT AND CONTENT OF APPLICATION FOR APPROVAL FOR THERMAL ANNEALING OF REACTOR PRESSURE VESSELS

During the 401st meeting of the Advisory Committee on Reactor Safeguards, September 9-10, 1993, we discussed the subject proposed rules and draft regulatory guide. Our Subcommittee on Materials and Metallurgy reviewed these matters in detail at a meeting on August 16, 1993. During these meetings, we had the benefit of discussions with representatives of the NRC staff. We also had the benefit of the document referenced.

The need for these proposed rules and the draft guide was, in part, highlighted during the evaluation of the integrity of the Yankee Nuclear Power Station's reactor pressure vessel. We believe these rules and this guide should prove useful to the licensees and the NRC staff and recommend that they be issued for public comment. We would like an opportunity to review the proposed final version of these rules and guide after the public comments have been reconciled and before publication.

Additional comments by ACRS Members Ivan Catton and William J. Lindblad are presented below.

Sincerely,

J. Ernest Wilkins, Jr.
Chairman

Additional Comments By ACRS Members Ivan Catton and William J. Lindblad

Although we agree with the essence of the above letter, we oppose the elimination of the provision in Appendix H which currently permits a reduction of testing in Integrated Surveillance Programs where "initial results agree with predictions." The licensee's program is, after all, subject to staff approval on a "case-by-case basis." Licensees should have some flexibility in scheduling when they actually test specimens. This does not mean that specimens would not be irradiated.

Reference:

Memorandum dated August 20, 1993, from Allen L. Hiser, Jr., Office of Nuclear Regulatory Research, for Elpidio G. Igne, ACRS, Subject: Response to Request at ACRS Subcommittee Meeting, with the following:

- a. Federal Register Notice for Proposed Rule, 10 CFR Part 50, "Fracture Toughness Requirements for Light Water Reactor Pressure Vessels"
- b. A proposed rule (10 CFR 50.66) on thermal annealing of the reactor pressure vessel, "Requirements for Thermal Annealing of the Reactor Pressure Vessel"
- c. Amendments to 10 CFR Part 50 Appendix G, "Fracture Toughness Requirements"
- d. Amendments to 10 CFR Part 50 Appendix H, "Reactor Vessel Material Surveillance Program Requirements"
- e. A draft regulatory guide (DG-1027), "Format and Content of Application for Approval for Thermal Annealing of Reactor Pressure Vessels"

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

October 14, 1993

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

Dear Chairman Selin:

SUBJECT: PROPOSED FINAL AMENDMENTS TO 10 CFR PART 55 ON RENEWAL OF
LICENSES AND REQUALIFICATION REQUIREMENTS FOR LICENSED
OPERATORS

During the 402nd meeting of the Advisory Committee on Reactor Safeguards, October 7-8, 1993, we reviewed the NRC staff's proposed final amendments to 10 CFR Part 55, Operators' Licenses. During this meeting, we had the benefit of discussions with representatives of the NRC staff. We also had the benefit of the document referenced.

These proposed amendments would revise the current requalification regulations for licensed operators by eliminating the present requirements that they pass a requalification written examination and operating test administered by the NRC during their six-year license term. Licensed operators would continue to be required to pass the biennial requalification written examination and annual operating test administered by their plant training organizations. As part of the proposed rule change, licensees would be required to submit, upon request, their requalification written examinations or operating tests for NRC review. The staff points out that these changes will allow the redirection of NRC license examiner resources so that the examiners will be able to perform more comprehensive inspections of licensees' operator requalification programs.

We believe that these proposed amendments to 10 CFR Part 55 will be beneficial and recommend their adoption.

Sincerely,

J. Ernest Wilkins, Jr.
Chairman

Reference:

Commission Paper, Draft, from James M. Taylor, NRC Executive Director for Operations, for the Commissioners, Subject: Final Amendments to 10 CFR Part 55 on Renewal of Licenses and Requalification Requirements for Licensed Operators, transmitted by memorandum dated October 4, 1993, from Bill M. Morris, NRC, to John T. Larkins, ACRS

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

October 14, 1993

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

Dear Mr. Taylor:

SUBJECT: PROPOSED RULE AND DRAFT REGULATORY GUIDE TO ADDRESS
RESOLUTION OF GENERIC ISSUE 23, "REACTOR COOLANT PUMP
SEAL FAILURE"

During the 402nd meeting of the Advisory Committee on Reactor Safeguards, October 7-8, 1993, we reviewed a proposed rule and draft regulatory guide developed for resolving Generic Issue 23, "Reactor Coolant Pump Seal Failure." Our Subcommittee on Decay Heat Removal Systems met on October 5, 1993, to review this matter. During this review, we had discussions with representatives of the NRC staff, Northeast Utilities, and NUMARC. We also had the benefit of the referenced documents.

We do not believe that the proposed rule and draft regulatory guide should be issued for public comment in their present form. There are several aspects of the rulemaking package that need further attention. For example, the proposed rule is too deterministic and does not permit consideration of the wide variation in risk from pump seal LOCA among the different plants. We believe that the proposed rule should be revised to permit the use of a risk-based approach. Furthermore, the provision in the regulatory guide that either seal cooling be restored within 10 minutes or a seal leakage rate of 480 GPM per reactor coolant pump be assumed appears to be overly conservative. This provision will have a serious impact on the coping analyses already performed to meet the strictures of the Station Blackout Rule.

There is clearly a need for further dialogue among the NRC staff, the nuclear industry, and the ACRS. We wish to be kept informed regarding the resolution of this matter.

Sincerely,

A handwritten signature in cursive script, reading "J. Ernest Wilkins, Jr.", is positioned above the printed name.

J. Ernest Wilkins, Jr.
Chairman

Mr. James M. Taylor

2

October 14, 1993

References:

Memorandum dated August 3, 1993, from C. J. Heltemes, Jr., Office of Nuclear Regulatory Research, to J. Ernest Wilkins, Jr., ACRS Chairman, Subject: Proposed Rule for GI-23, "Reactor Coolant Pump Seal Failure," with following enclosures

- (1) Draft NUREG-1483, "Regulatory Analysis Supporting Proposed Rule on Reactor Coolant Pump Seals"
- (2) Proposed Federal Register notice
- (3) Draft Regulatory Guide, DG-1008, Revision 1, "Reactor Coolant Pump Seals."

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

December 10, 1993

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

Dear Chairman Selin:

SUBJECT: PROPOSED AMENDMENTS TO 10 CFR PART 73 TO PROTECT AGAINST
MALEVOLENT USE OF VEHICLES AT NUCLEAR POWER PLANTS

During the 403rd meeting of the Advisory Committee on Reactor Safeguards, November 4-6, 1993, we discussed SECY-93-270, which contains proposed amendments to 10 CFR Part 73 to protect against malevolent use of vehicles at nuclear power plants. Our Subcommittee on Safeguards and Security reviewed this matter during a meeting on November 3, 1993. During this review we had the benefit of discussions with representatives of the NRC staff. We also had the benefit of the document referenced.

We do not support the proposal to go ahead with expedited rule-making, and regret that the issue came to us so late in the process that it is awkward to apply brakes now. But it is never too late.

The stated reason for enhancing the defenses of nuclear power plants against attack through vehicle-borne people or explosives (the staff interprets the word vehicle in the narrow sense, as car or truck) is that the attack on the World Trade Center and the unplanned intrusion at Three Mile Island Unit 1 provide bases for increasing both the Design-Basis Threat and the "Actual Threat." The latter is a euphemism for the best intelligence information available to NRC. We do not believe that either increase is justified by the facts. It is particularly disturbing that the proposed amendments and consequent backfits are on a fast track, lacking the customary analysis, and that the Commission has apparently endorsed this approach.

A threat is always a function of both the level of potential harm to the public and its probability--no matter how challenging it may be to estimate the latter it is always possible. Indeed, an effort to do so can serve to enforce clear thinking. The staff has made no effort to estimate the likelihood of a malevolent intrusion, either before or after the two events cited, but has simply asserted that the risk has increased, and that the increase alone justifies the imposition of new requirements.

One of the more praiseworthy trends in risk management in recent years has been toward effective use of probability as a tool in regulation, whether through risk-based maintenance, cost-benefit analysis of backfits, promulgation of safety goals, or other mechanisms. The Commission has repeatedly endorsed this trend, and it is disheartening to see it so blatantly ignored in this case.

Lest there be a misunderstanding, we do not suggest for a moment that there is no risk, only that there is no basis for the conclusion that it has recently and substantially increased.

The staff also asserts that it is newly concerned about timely warning of the accumulation of explosive materials in sufficient quantity to support an attack. We find that puzzling. All truck bombings for which the explosive material is known to any of us have been conducted with ANFO (ammonium nitrate/fuel oil), and the intelligence agencies have never been able to track the flow. Not only is there prodigious national use of this explosive for blasting (in the order of a megaton per year), but other normal uses of these base materials far exceed this. None of us know of any recent change in the difficulty of tracking this kind of explosive, nor did the staff suggest any.

We believe that the imposition of new requirements to counter the threat of vehicular attack should follow the usual orderly path of analysis for both cost and effectiveness. The latter should include a realistic assessment of the threat, including probabilities. Of course this is hard, but it is not impossible. And it depends on the thoughtful use of the best available intelligence data.

We are unconvinced of the need to rush into rulemaking. If there is special information unknown to any of us, leading to the conclusion that the threat has truly changed substantially for the worse, then there may not be adequate time for rulemaking, and the Commission should invoke its emergency authority. There is little to be said for the present course, too slow if the need is urgent, too fast if it is not.

Additional comments by ACRS Members Peter R. Davis, Carlyle Michelson, William J. Shack, and Charles J. Wylie are presented below.

Sincerely,

A handwritten signature in dark ink, reading "J. Ernest Wilkins, Jr." in a cursive style.

J. Ernest Wilkins, Jr.
Chairman

Additional Comments by ACRS Members Peter R. Davis, Carlyle Michelson, William J. Shack, and Charles J. Wylie

We support the staff recommendation stated in SECY-93-270. It is our view that the proposed rule represents a prudent and effective step toward enhancing public health and safety.

Reference:

SECY-93-270, Proposed Amendments to 10 CFR Part 73 to Protect Against Malevolent Use of Vehicles at Nuclear Power Plants, dated September 29, 1993

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

April 22, 1993

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

Dear Chairman Selin:

SUBJECT: DEFINITION OF A LARGE RELEASE FOR USE WITH SAFETY GOAL
POLICY

During the 396th meeting of the Advisory Committee on Reactor Safeguards, April 15-17, 1993, we discussed the staff's recommendations in regard to the definition of a large release related to the implementation of the Commission's Safety Goal Policy. During this meeting, we had the benefit of discussions with members of the NRC staff and of the document referenced.

In the draft Commission paper and in the presentation to the Committee, the staff expressed its belief that the development of the definition of a large release is no longer practical or useful and, therefore, it is requesting Commission approval to terminate efforts in this area. We believe the staff has made a conscientious effort with this activity and we agree with its basic conclusions. Our views are as follows:

1. A large release definition would either represent a replacement for the existing safety goals or, if made consistent with the quantitative health objectives (QHOs), would be redundant and unnecessary.
2. New guidelines being developed for implementing the Safety Goal Policy within regulatory analysis and issue prioritization processes adequately meet the originally perceived need for a large release component of the safety goals. These utilize a core damage frequency (CDF) and a conditional containment failure probability (CCFP).
3. Plant performance objectives, viz $CDF \leq 10^{-4}$ and $CCFP \leq 0.1$, provide an easily understandable and adequate surrogate for the QHOs and provide quantitative prioritization for two basic aspects of defense in depth (prevention and mitigation). These could help ensure that a plant does not end up with great core protection but marginal containment performance.

The Honorable Ivan Selin

2

April 22, 1993

We support the recommendation that the Commission approve the staff's proposal to terminate its effort to develop a definition of a large release.

Sincerely,

A handwritten signature in cursive script that reads "Paul Shewmon". The letters are fluidly connected, with a large initial "P" and a long, sweeping underline for the "mon" part.

Paul Shewmon
Chairman

Reference:

Memorandum dated March 11, 1993, from Warren Minners, Director, RES/DSIR, for John T. Larkins, Acting Executive Director, ACRS, Subject: ACRS Review of Draft Commission Paper on Large Release Determination, w/Enclosure

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

June 25, 1993

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

Dear Chairman Selin:

In our report dated May 26, 1993, "Staff Approach for Assessing the Consistency of the Present Regulations with Respect to the Commission's Safety Goals," there was an unfortunate error in wording, which may have caused some confusion. In the last paragraph the words, "level of safety" should be replaced by "level of risk." We were speaking of safety levels better than the safety goals, therefore, lower risk. We regret the error. We are transmitting a revised copy of the report to you.

Sincerely,

A handwritten signature in cursive script, reading "J. Ernest Wilkins, Jr.", is positioned below the word "Sincerely,".

J. Ernest Wilkins, Jr.
Chairman

Enclosure:

ACRS report dated May 26, 1993, from Paul Shewmon, ACRS Chairman, to The Honorable Ivan Selin, NRC Chairman, Revised June 16, 1993, Subject: Staff Approach for Assessing the Consistency of the Present Regulations With Respect to the Commission's Safety Goals

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

May 26, 1993

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

Dear Chairman Selin:

**SUBJECT: STAFF APPROACH FOR ASSESSING THE CONSISTENCY OF THE
PRESENT REGULATIONS WITH RESPECT TO THE COMMISSION'S
SAFETY GOALS**

During the 397th meeting of the Advisory Committee on Reactor Safeguards, May 13-15, 1993, we discussed a draft Commission paper regarding the staff's proposed approach for assessing the consistency of present regulations with respect to the Commission's safety goals. During this meeting, we had the benefit of discussions with representatives of the staff.

In a Staff Requirements Memorandum (SRM) dated June 15, 1990, the Commission requested that the staff develop a plan "for assessing the consistency of our regulations with the safety goals." This is an effort that the Committee has recommended in several reports, and continues to endorse.

In its presentation, the staff provided a conclusion that a specific new program is not necessary to respond to the SRM. The staff contends that existing programs, in the areas noted below, are sufficient to make the desired assessment:

1. Elimination of Requirements Marginal to Safety
2. IPE/IPEEE Data Base Insights
3. Other ongoing activities that include:
 - The Regulatory Review Group
 - Generic Safety Issue evaluations
 - AEOD evaluations of operational events and data
 - NRR inspection reports
 - Accident Sequence Precursor studies

We believe that these existing programs can provide input into the subject program, but are not by themselves responsive to the SRM. We recommend that a directed effort be undertaken to make the

assessments requested in the SRM. A first step should be to develop an assessment strategy to make use of the IPE/IPEEE results and other appropriate PRA results to establish the existing level of safety that has resulted from compliance with the body of current regulations, to be compared with the safety goals.

The facts that the IPEs are essentially Level 2 PRAs and do not evaluate risk directly, and that seismic and fire events in IPEEEs are not necessarily evaluated probabilistically, are formidable barriers to their use for assessing the consistency of the present regulations with the safety goals. Nevertheless, these and other existing PRAs are the best available information for such an assessment. We recommend that the assessment strategy include the development of surrogates for the safety goals, expressed in terms of core damage probability and conditional containment failure probability - the outputs of the IPE. We believe that bounding, site-independent surrogates can be developed because, for high source terms, the conditional mean individual risk of early fatalities approaches a limit of about 0.1, and the conditional mean individual risk for latent fatalities approaches a limit of about 0.01. These limits result from the probability that the wind will blow in a given direction.

It is entirely possible that the outcome of such an assessment will reveal that the level of risk resulting from compliance with the body of existing regulations is below the safety goal levels of risk. Such a finding would have significant implications. It is important that such a determination be made.

Sincerely,

Paul Shewmon
Chairman

Reference:

1. Memorandum dated April 18, 1993, from C. J. Heltemes, Office of Nuclear Regulatory Research, for John T. Larkins, ACRS, Subject: Staff Approach for Assessing the Consistency of the Present Regulations with Respect to the Commission's Safety Goals, with attachments:
 - a. SRM dated June 15, 1990, Subject: SECY-89-102, Subject: Implementation of the Safety Goals
 - b. Draft SECY paper for the Commissioners from James M. Taylor, EDO, Subject: Staff Approach for Assessing the Consistency of the Present Regulations with Respect to the Commission's Safety Goals (Predecisional)

The Honorable Ivan Selin

3

May 26, 1993

2. ACRS Report dated April 12, 1988, from W. Kerr, ACRS Chairman, to The Honorable Lando W. Zech, Jr., NRC Chairman, Subject: Program to Implement the Safety Goal Policy -- ACRS Comments

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

January 28, 1993

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 (ACRS) has reported to the Congress each year on the Safety Research Program of the Nuclear Regulatory Commission (NRC). In our December 18, 1986, letter to the Congress, we proposed to provide reports on specific issues rather than one all-inclusive report, as we had provided before 1986. Accordingly, enclosed are copies of reports that we have provided to the NRC during the past year on matters related to the agency's research program.

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

Sincerely,

Paul Shewmon

Paul Shewmon
Chairman, ACRS

*Enclosures:

1. Report from David A. Ward, ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: Requirements for Full-Height, Full-Pressure Integral System Testing of the Westinghouse AP600 Passive Plant Design, March 10, 1992
2. Report from David A. Ward, ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: Full-Height, Full-Pressure Integral System Testing for the Westinghouse AP600 Passive Plant Design, April 6, 1992
3. Report from David A. Ward, ACRS Chairman, to James M. Taylor, Executive Director for Operations, NRC, Subject: Evaluation of the Risks During Shutdown and Low-Power Operations for U.S. Nuclear Power Plants, April 9, 1992

4. Report from David A. Ward, ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: Testing and Analysis Programs in Support of the Simplified Boiling Water Reactor Design Certification, June 10, 1992
5. Report from Paul Shewmon, ACRS Acting Chairman, to Ivan Selin, NRC Chairman, Subject: Integral System and Separate Effects Testing in Support of the Westinghouse AP600 Plant Design Certification, July 17, 1992
6. Report from David A. Ward, ACRS Chairman, to James M. Taylor, Executive Director for Operations, NRC, Subject: Proposed Resolution of Generic Safety Issue 106, "Piping and the Use of Highly Combustible Gases in Vital Areas," August 14, 1992
7. Report from David A. Ward, ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: Severe Accident Research Program Plan, August 18, 1992
8. Report from David A. Ward, ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: Digital Instrumentation and Control System Reliability, September 16, 1992
9. Report from Paul Shewmon, ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: Environmental Qualification for Digital Instrumentation and Control Systems, November 12, 1992

*For Items 1 through 9, see NUREG-1125, Volume 14, 4/93.

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

January 28, 1993

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

Dear Mr. President:

In accordance with the requirements of Section 29 of the Atomic Energy Act of 1954, as amended by Section 5 of Public Law 95-209, the Advisory Committee on Reactor Safeguards (ACRS) has reported to the Congress each year on the Safety Research Program of the Nuclear Regulatory Commission (NRC). In our December 18, 1986, letter to the Congress, we proposed to provide reports on specific issues rather than one all-inclusive report, as we had provided before 1986. Accordingly, enclosed are copies of reports that we have provided to the NRC during the past year on matters related to the agency's research program.

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

Sincerely,

Paul Shewmon

Paul Shewmon
Chairman, ACRS

*Enclosures:

1. Report from David A. Ward, ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: Requirements for Full-Height, Full-Pressure Integral System Testing of the Westinghouse AP600 Passive Plant Design, March 10, 1992
2. Report from David A. Ward, ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: Full-Height, Full-Pressure Integral System Testing for the Westinghouse AP600 Passive Plant Design, April 6, 1992
3. Report from David A. Ward, ACRS Chairman, to James M. Taylor, Executive Director for Operations, NRC, Subject: Evaluation of the Risks During Shutdown and Low-Power Operations for U.S. Nuclear Power Plants, April 9, 1992

4. Report from David A. Ward, ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: Testing and Analysis Programs in Support of the Simplified Boiling Water Reactor Design Certification, June 10, 1992
5. Report from Paul Shewmon, ACRS Acting Chairman, to Ivan Selin, NRC Chairman, Subject: Integral System and Separate Effects Testing in Support of the Westinghouse AP600 Plant Design Certification, July 17, 1992
6. Report from David A. Ward, ACRS Chairman, to James M. Taylor, Executive Director for Operations, NRC, Subject: Proposed Resolution of Generic Safety Issue 106, "Piping and the Use of Highly Combustible Gases in Vital Areas," August 14, 1992
7. Report from David A. Ward, ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: Severe Accident Research Program Plan, August 18, 1992
8. Report from David A. Ward, ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: Digital Instrumentation and Control System Reliability, September 16, 1992
9. Report from Paul Shewmon, ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: Environmental Qualification for Digital Instrumentation and Control Systems, November 12, 1992

*For Items 1 through 9, see NUREG-1125, Volume 14, 4/93.

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

June 18, 1993

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

Dear Mr. Taylor:

SUBJECT: PUBLIC COMMENTS ON PROPOSED RULE ON ALWR SEVERE ACCIDENT PERFORMANCE

During the 398th meeting of the Advisory Committee on Reactor Safeguards, June 10-11, 1993, we discussed with members of the staff public comments received on the Advance Notice of Proposed Rulemaking (ANPR) on ALWR Severe Accident Performance. We had the benefit of the documents referenced.

It is our understanding that the staff's proposed approach for proceeding with rulemaking involves the following four elements:

1. Continuing discussions with ACRS concerning a potential generic rule,
2. Delaying a final decision on implementation of the rule until after final safety evaluation reports are issued for the Advanced Boiling Water Reactor (ABWR) and the CE System 80+,
3. Coordinating the efforts of drafting a generic rule and the design certification rules for the ABWR and the CE System 80+ to ensure consistency, and
4. Following the reviews of the evolutionary and passive reactor designs to ensure consistency of the draft rule with these reviews.

We agree with this approach.

In our reports on this subject dated May 17, 1991 and May 14, 1992, we developed and subsequently endorsed what is designated as Alternative 3 in the ANPR. We continue to recommend this alternative.

June 18, 1993

For your further consideration, we recommend that your approach accommodate the following:

1. The amended regulations should not be so prescriptive as to preclude the use of a design feature which substantially reduces the challenge(s) to the containment. For example, the approach should not require accommodation of large amounts of hydrogen generation if a design change (such as different core materials) precludes significant hydrogen generation,
2. The recognition of passive design features to cope with some phenomena, e.g., a large volume-high strength containment, and
3. Consideration for dealing with combinations of containment loads from severe accident phenomena, e.g., steam explosions and hydrogen combustion/detonation.

We expect to have further discussions with the staff on this matter.

Sincerely,

J. Ernest Wilkins, Jr.
Chairman

References:

1. Memorandum dated May 14, 1993, from Warren Minners, Office of Nuclear Regulatory Research, for John T. Larkins, Advisory Committee on Reactor Safeguards, Subject: Summary of Public Comments on Proposed Rule on ALWR Severe Accident Performance - 57 FR 44513 (Predecisional Draft Commission Paper Attached)
2. Report dated May 17, 1991, from David A. Ward, Chairman, ACRS, to Kenneth M. Carr, Chairman, NRC, Subject: Proposed Criteria to Accommodate Severe Accidents in Containment Design
3. Report dated May 14, 1992, from David A. Ward, Chairman, ACRS, to James M. Taylor, Executive Director for Operations, NRC, Subject: Advance Notice of Proposed Rulemaking on Severe Accident Plant Performance Criteria for Future LWRs

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

December 16, 1993

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

Dear Mr. Taylor:

SUBJECT: INDIVIDUAL PLANT EXAMINATION PROGRAM

During the 404th meeting of the Advisory Committee on Reactor Safeguards, December 9-11, 1993, we discussed the status of the Individual Plant Examination (IPE) program and some aspects of the resolution of Unresolved Safety Issues (USIs) and Generic Safety Issues (GSIs) by the IPE and the Individual Plant Examination of External Events (IPEEE) programs. These matters were also discussed during a meeting of our Subcommittee on Individual Plant Examinations on November 18, 1993. During this review, we had the benefit of discussions with representatives of the NRC staff. We also had the benefit of the documents referenced.

We share the conclusion expressed by the staff that the IPE program appears to have exceeded expectations. We are particularly encouraged that (1) licensees are actually using their IPE results to effect safety improvements at their plants and (2) most, if not all, licensees plan to maintain their IPEs as an ongoing current assessment of plant risks (although not required), and to use the results as an important adjunct in making decisions with potential safety implications. We also note that the definition of the database structure to be used in the collection and correlation of IPE/IPEEE program results has received careful attention early in the program. Both the format for entering results into the database and flexible retrieval capabilities will be provided for users. A users handbook is scheduled to be available in mid-1994. We do have two concerns about the IPE process that we will provide later in this letter.

With respect to the IPEEE process, we are in general agreement with the staff approach. We would recommend, however, that the staff consider the possibility of developing some method for converting the qualitative approaches for evaluating external events (such as the Fire Induced Vulnerability Evaluation and the Seismic Margin Assessment methodologies) into quantitative equivalents. This would facilitate determinations of relative significance, provide a more definitive framework for decisionmaking, and aid in an

overall assessment of the status of the population of plants with respect to the safety goals.

Regarding the Accident Management program, we expect to comment when the program is more nearly complete. We are favorably impressed, however, with the extensive and constructive interaction between the staff and industry during the development of accident management strategies. We encourage a continuation of this interaction.

Returning to the IPE process, we have two concerns, as follows:

1. In our limited review of several IPE results, we were perplexed by the wide variation in reported values for conditional containment failure probability (CCFP). The values ranged from a few percent to 80 percent, and this was based on only a few IPE samples. Since in many cases neither public risks nor large release probabilities are provided (nor were they required) in the IPEs, it is difficult to decide how to determine the existence of a containment performance vulnerability. Part of the large CCFP range appears to be related to a variation in the definition of "containment failure." For example, it appears that different views have been taken on whether basemat meltthrough, deliberate venting, or interfacing system LOCA events constitute containment failure. Furthermore, the mode and timing of containment failure, as well as the precursor events postulated to occur during a core melt sequence, can cause vast variations in the estimated atmospheric source term (not computed in many of the IPEs). For example, an early overpressure failure in a PWR is obviously of much greater concern in terms of adverse public impact than a late containment basemat meltthrough. Yet, both failure modes may show up in the IPEs as equivalent contributors to CCFP.

As a result of the inconsistent definition of containment failure and the potential for an exceedingly wide variation in the source term resulting from different containment failures, it is not clear how the staff can draw any meaningful conclusions regarding containment vulnerabilities for an individual plant. We recommend that the staff give this matter additional consideration and try to establish a framework for evaluating containment performance for severe accidents from the IPE information. Part of this might include the formulation of an impact index to describe the relative risk significance of various modes of containment failure described in the IPEs based on equivalent containment failure parameters from the NUREG-1150 results as well as the results from other Level III PRAs.

2. We are concerned that the resolution of safety issues (USIs/GSIs) by the IPE process is not being tracked and evaluated by the staff. We agree, as we have stated in the past, that the IPE/IPEEE process is an appropriate mechanism for the resolution of those USIs/GSIs that appear to be highly plant-specific. However, we were informed that the IPE reviews are not to be focused on the treatment and corresponding results for the USIs/GSIs which are expected to be included in the IPEs. (This position is consistent with the information contained in your letter to the ACRS Chairman dated September 22, 1993.) We are concerned that inadequate or incorrect models and assumptions might be used in the treatment of the USIs/GSIs. This can obscure the significance of the USIs/GSIs for a particular plant and would not be discovered because of the incomplete review process. We note in this regard that the latest IPE Review Guidance Document does not specify review requirements for USIs/GSIs. The staff previously determined that the USIs/GSIs had potential safety significance (or they would not be safety issues), based on the results of staff research and assessment. It therefore seems important that the IPEs be reviewed specifically to ensure that (a) the USIs/GSIs have been considered by the IPE assessments, and (b) the models and assumptions associated with the treatment of the USIs/GSIs in the IPE be consistent with the staff's understanding of these issues based on the evaluations performed to establish the significance of the USIs/GSIs. It seems to us that this review could be facilitated by incorporating specific USIs/GSIs into the database being created by Brookhaven National Laboratory from the IPE results.

We look forward to continued interaction with the staff on these important topics.

Sincerely,

J. Ernest Wilkins, Jr.
Chairman

References:

1. U.S. Nuclear Regulatory Commission, Generic Letter No. 88-20, November 23, 1988, "Individual Plant Examination for Severe Accident Vulnerabilities - 10 CFR §50.54(f)"
2. U.S. Nuclear Regulatory Commission, Generic Letter No. 88-20, Supplement 4, June 28, 1991, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10 CFR 50.54(f)"

3. Letter dated September 22, 1993, from Mr. James M. Taylor, Executive Director for Operations, for Dr. J. Ernest Wilkins, Jr., ACRS Chairman, Subject: ACRS Letter Dated August 11, 1993: Proposed Resolution of Generic Issue 143, "Availability of Chilled Water System and Room Cooling"
4. Memorandum dated November 18, 1993, from Warren Minners, Office of Nuclear Regulatory Research, for John Larkins, ACRS, Subject: IPE Review Guidance Document (Draft Predecisional)

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