NUREG-1125
Volume 19

A Compilation of
Reports of
The Advisory
Committee on
Reactor
Safeguards

1997 Annual

U. S. Nuclear Regulatory
Commission

April 1998

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MASTER

April 1998

ABSTRACT

This compilation contains 67 ACRS reports submitted to the Commission, or to the Executive Director for Operations, during calendar year 1997. It also includes a report to the Congress on the NRC Safety Research Program. All reports have been made available to the public through the NRC Public Document Room, the U. S. Library of Congress, and the Internet at <http://www.nrc.gov/ACRSACNW>. The reports are categorized by the most appropriate generic subject area and by chronological order within the 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 1997. 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
15	Calendar Year 1993
16	Calendar Year 1994
17	Calendar Year 1995
18	Calendar Year 1996

ACRS MEMBERSHIP (1997)

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Sandia National Laboratories

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Massachusetts Institute of Technology

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GPU Nuclear Corporation

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University of Tennessee

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Dr. Don W. Miller, Professor
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Argonne National Laboratory

Dr. Robert E. Uhrig, Professor
University of Tennessee

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

March 17, 1997

Mr. L. Joseph Callan
Executive Director for Operations
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Mr. Callan:

SUBJECT: PUBLICATION OF PROPOSED JOURNAL ARTICLE CONTAINING ROSA
TEST DATA

In a December 13, 1996 letter, Mr. James M. Taylor (then Executive Director for Operations) requested that the ACRS provide an independent review of the technical merits of a Westinghouse Electric Corporation concern that the data from some of the ROSA-V tests included in a proposed journal article jointly authored by NRC and Japan Atomic Energy Research Institute personnel when combined with other publicly available information could permit a competitor to deduce proprietary design details through "reverse engineering."

During our 439th meeting, March 6-8, 1997, we reviewed the merits of this concern. Our Subcommittee on Thermal-Hydraulic Phenomena also reviewed this issue during a meeting on February 19, 1997. During this review, we had the benefit of discussions with representatives of the NRC staff and Westinghouse Electric Corporation. We also had the benefit of the documents referenced.

During the February 19, 1997 Subcommittee meeting, representatives of Westinghouse stated that their concern was somewhat broader than that posed to the Committee by Mr. Taylor. Their broader concern was that a third party could use information as presented in the proposed article along with other available information to provide credibility to a competing design and thereby undermine the competitive international marketing position of Westinghouse.

We recognize that there are elements of this issue that will not be captured by a strictly technical review. These could include, for example, NRC needs for (1) sufficient public disclosure of its safety case, (2) independence from the vendors, and (3) uninhibited future publication of data from other test facilities. Nevertheless, in our review, we have chosen to focus on two technical questions associated with the broader concern:

- (1) Is there sufficient information available in the proposed article (in its current form) that could be used along with

other information in the public domain by a knowledgeable third party to develop a technically defensible thermal-hydraulic computer mockup of the ROSA-V facility?

- (2) If so, can the data in the form currently proposed to be published in the article be used along with a computer model to provide credibility to a competing design for which the performance of the passive systems is demonstrated via the computer model?

It is our opinion that a technically defensible computer model of the ROSA-V facility could be developed by a third party from the information in the proposed article when combined with other available information in the public domain, and that such a model could be an important factor in providing credibility to a competing passive design. A claim could be made that the design is based on an NRC-approved design. In our opinion, there is significant merit to the concerns of Westinghouse and these concerns ought to be given appropriate consideration in any decision regarding the form of data presentation in the proposed article.

Sincerely,

R. L. Seale
Chairman

References

1. Letter dated December 13, 1996, from James M. Taylor, Executive Director for Operations, NRC, to Thomas S. Kress, Chairman, ACRS, Subject: ACRS Review of ROSA and APEX Data Prior to Publication.
2. Proposed journal article, "Implications of the Rig of Safety Assessment/AP600 High- and Intermediate-Pressure Test Results," by Louis M. Shotkin, Nuclear Regulatory Commission, and Yutaka Kukita, Japan Atomic Energy Research Institute (**Review for Public Release Pending**).
3. Letter dated February 4, 1997, from Brian A. McIntyre, Westinghouse Electric Corporation, to Robert Seale, Chairman, ACRS, Subject: ACRS Review of the Proposed Publication of a Journal Article Concerning AP600 Test Results.
4. Proposed journal article, "Core Markup Tank Behavior Observed During the ROSA-AP600 Experiments," by T. Yotomot, Japan Atomic Energy Research Institute, et al. (**Review for Public Release Pending**).
5. Proposed paper, "ROSA-AP600 Experiment Simulating Steam Generator Tube Rupture Transient," by H. Nakamura and Y. Kukita, Japan Atomic Energy Research Institute, for the 2nd

International Topical Meeting on Advance Reactors Safety, Orlando, Florida, June 1-4, 1997.

6. Proposed paper, "Analysis of Wall Heat Capacity Effect on Core Makeup Tank Drain-Down Behavior in ROSA/AP600 Experiment," by Masaya Kondo, Japan Atomic Energy Research Institute, et al., for the 2nd International Topical Meeting on Advance Reactors Safety, Orlando, Florida, June 1-4, 1997.
7. American Nuclear Society proceedings paper, "Passive Residual Heat Removal System Heat Exchanger Characterization During Simulated Station Blackout," by Owen Stevens and Jose N. Reyes, Jr., Oregon State University, from the 1996 National Heat Transfer Conference, Houston, Texas, August 3-6, 1996.
8. American Nuclear Society proceedings paper, "Evaluation of the APEX Break Flow Measurement System During Subcooled Depressurization," by David A. Pimentel and Jose N. Reyes, Jr., Oregon State University, from the 1996 National Heat Transfer Conference, Houston, Texas, August 3-6, 1996.
9. International Atomic Energy Agency proceedings paper, "SPES-2. AP600 Integral Systems Test Results," by L. E. Conway and R. Hundal, Westinghouse Electric Corporation, Italy, May 1995.
10. Proceedings paper, "Comparison of the SPES-2 Pre-Test Predictions and AP600 Plant Calculations Using RELAP5/MOD3," by A. Alemberti, C. Frepoli, and G. Graziosi, ANSALDO Nuclear Division (Copies available from the NRC Office of Nuclear Reactor Regulation).
11. Proceedings paper, "SPES-2 Cold Leg Break Experiments: Scaling Approach for Decay Power, Heat Losses Compensation and Metal Heat Release," by A. Alemberti, C. Frepoli, and G. Graziosi, ANSALDO Nuclear Division (Copies available from the NRC Office of Nuclear Reactor Regulation).
12. Proceedings paper, "SPES-2, The Full-Height, Full-Pressure, Integral System AP600 Test Facility," by M. Bacchiani, SIET S.P.A., et al., Twenty-Second Water Reactor Safety Information Meeting, Bethesda, Maryland, October 24-26, 1994.
13. Proceedings paper, "SPES-2 RELAP5/MOD3 Noding and 1 Inch Cold Leg Break Test S00401," by A. Alemberti, C. Frepoli, and G. Graziosi, ANSALDO Nuclear Division, Twenty-Second Water Reactor Safety Information Meeting, Bethesda, Maryland, October 24-26, 1994.
14. Abstract of thesis, "Characterization of the Advanced Plant Experiment (APEX) Passive Residual Heat Removal System Heat Exchanger," by Owen Stevens, Oregon State University, presented June 7, 1996.
15. Abstract of thesis, "Two-Phase Fluid Break Flow Measurements and Scaling in the Advanced Plant Experiment (APEX)," by David Alan Pimentel, Oregon State University, presented on May 9, 1996.

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

March 19, 1997

Mr. L. Joseph Callan
Executive Director for Operations
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555-0001

Dear Mr. Callan:

SUBJECT: NRC OFFICE OF NUCLEAR REGULATORY RESEARCH PROGRAM TO
DEMONSTRATE THE ADEQUACY OF THE RELAP5/MOD3 CODE TO
ANALYZE AP600 PASSIVE PLANT BEHAVIOR

During the 439th meeting of the Advisory Committee on Reactor Safeguards, March 6-8, 1997, we completed our review of the NRC Office of Nuclear Regulatory Research (RES) program to demonstrate the adequacy of the RELAP5/MOD3 code for analyzing the behavior of the Westinghouse AP600 passive plant design. Our Subcommittee on Thermal-Hydraulic Phenomena met to consider this matter on February 22-23, 1996 and February 12-14, 1997. We previously commented on our initial review of this issue in a March 19, 1996 letter to the Executive Director for Operations. During this review, we had the benefit of discussions with representatives of the NRC staff, and we also had the benefit of the documents referenced.

Scaling

The staff sponsored two scaling analyses to show that data obtained from several integral and separate effects test facilities are applicable to AP600 and can be used to validate RELAP5/MOD3.2.1.2 for assessing the ability of the AP600 design to address design basis accidents (DBAs). Both sets of analyses demonstrated that the data available are appropriate for validating the code. We agree with the conclusions of these analyses and compliment the staff and its contractors on a job well done. A final summary report consolidating the results of these two analyses should be written to document the conclusion that the data are sufficiently complete and also to note how the tests were used for code validation.

Code Adequacy

The code documentation made available to us was out of date. As a result, our conclusions about code adequacy are based partly on the additional information presented at the meetings. Based on our review of this information, we believe that RELAP5/MOD3.2.1.2 is

adequate for evaluating Westinghouse AP600 DBAs. The RELAP5/MOD3.2.1.2 calculations for particular DBAs showed that sufficient margin exists in the AP600 design. The tests and the associated code predictions indicate that the core does not uncover under any of the tested DBA conditions. Documentation reflecting what was presented to us should be completed in a timely manner. We also believe that there is a need to develop more quantitative criteria than those used by RES for assessing the adequacy of a thermal-hydraulic code.

During our review, a number of deficiencies in the RELAP5/MOD3.2.1.2 code were identified but were determined not to significantly affect the code's capability to evaluate the AP600 emergency core cooling system behavior. Nevertheless, these deficiencies indicate that the code falls short of being a true best-estimate code. (These deficiencies are also generally found in the TRAC code.) The deficiencies identified include the following:

1. The critical flow model is a "work-around" that is appropriate only for this specific application.
2. The treatment of subcooled nucleate boiling in the core is poor.
3. The code does not model horizontal stratified flow very well.
4. The drift-flux correlations used in the code have known deficiencies at low pressure.
5. The code inadequately treats condensation and thermal stratification.
6. Treatment of the automatic depressurization system valves in both the code and in the experimental studies is overly simplistic.

There is a need to develop sound models in these areas for the future consolidated systems code, and we recommend that a modest effort be continued to also improve these areas in the RELAP5 code.

Inevitably, there will be uses of RELAP5/MOD3 other than demonstrating adequacy of the AP600 design. These uses could, for example, include studies ranging from loss of coolant accidents and transients to simulator fidelity evaluation. For some of these, it is sufficient to have a conservative analysis, whereas for others the results will need to be as realistic as possible. For any such additional uses of the code, the effects of the identified deficiencies need to be evaluated.

Modeling a nuclear power plant is a complex task that requires a mix of first principles combined with a great deal of physical insight and engineering judgment. As a result, personnel who are knowledgeable about the thermal-hydraulic behavior of single and two-phase flow as well as the use of computers must be available if sound judgments are to be made.

The thermal-hydraulics team assembled by the NRC for this exercise demonstrated a great deal of understanding relative to both the AP600 test results and the accident behavior of the nominal plant design. We believe that this work is excellent preparation for the current effort of the NRC staff to develop a new "consolidated" thermal-hydraulics code and augurs well for its success.

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

Sincerely,

R. L. Seale
Chairman

References:

1. U. S. Nuclear Regulatory Commission Report, NUREG/CR-5535, "RELAP5/MOD3 Code Manual," Volumes 1-7, August 1995.
2. Letter dated March 19, 1996, from T. S. Kress, Chairman, ACRS, to J. M. Taylor, Executive Director for Operations, NRC, Subject: NRC Staff Program on the Adequacy Assessment of the RELAP5/MOD3 Code for Simulation of AP600 Passive Plant Behavior.
3. Memorandum dated December 27, 1996, from M. W. Hodges, Office of Nuclear Regulatory Research, NRC, to J. T. Larkins, Executive Director, ACRS, Subject: Adequacy Assessment of the RELAP5 MOD3 Code to Analyze AP600 Passive Plant Behavior Following Postulated Small Break Loss-Of-Coolant Accidents, transmitting the following reports:
 - a. Idaho National Engineering Laboratory, Final Draft Report, INEL-96/0400, "Adequacy Evaluation of RELAP5/MOD3, Version 3.2.1.2 for Simulating AP600 Small Break Loss-of-Coolant Accidents," December 1996, with Appendix (Proprietary Information).
 - b. Idaho National Engineering Laboratory Report, INEL-96/0395, "Evaluation and Assessment of RELAP5/MOD3 Version 3.2.1.2 for Simulating the Long-Term Phase of Small Break Loss-of-Coolant Accidents in the AP600," October 1996 (Proprietary Information).

- c. Idaho National Engineering Laboratory, Final Draft Report, INEL-96/0440, "Applicability of Selected RELAP5/MOD3 Models and Correlations to AP600 Small Break Loss-of-Coolant Accidents," November 1996 (Proprietary Information).
 - d. Idaho National Engineering Laboratory, Final Draft Report, INEL-96/0117, "Top-Down Scaling Analyses of AP600 Integral Test Data in Support of RELAP5 Adequacy Assessment," December 1996 (Proprietary Information).
 - e. Idaho National Engineering Laboratory Report, INEL-94/0061, "Phenomena Identification and Ranking Tables for Westinghouse AP600 Small Break Loss-of-Coolant Accident, Main Steam Line Break, and Steam Generator Tube Rupture Scenarios," Revision 2, November 1996.
 - f. U. S. Nuclear Regulatory Commission, Draft Report, "Phenomenology Observed in the AP600 Integral Systems Test Programs Conducted in the ROSA-AP600, APEX, and SPES-2 Facilities," D. Bessette, Office of Nuclear Regulatory Research, NRC; M. DiMarzo, University of Maryland; P. Griffith, Massachusetts Institute of Technology, December 1996 (Proprietary Information).
 - g. U. S. Nuclear Regulatory Commission, Draft Report, "Screening Reactor Steam/Water Piping Systems for Water Hammer," P. Griffith, Consultant, November 1996.
4. Idaho National Engineering Laboratory, Draft Report, INEL-96/0040, "Top-Down Scaling Analysis Methodology for AP600 Integral Tests," S. Banerjee, et al., January 1997 (Proprietary Information).

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

June 17, 1997

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

Dear Chairman Jackson:

SUBJECT: PROPOSED STAFF POSITION REGARDING INCLUSION OF A
CONTAINMENT SPRAY SYSTEM IN THE AP600 DESIGN.

During the 442nd meeting of the Advisory Committee on Reactor Safeguards, June 11-14, 1997, we met with representatives of the NRC staff and the Westinghouse Electric Corporation to discuss the proposed staff position that the AP600 design should include a containment spray system or equivalent for accident management following a severe accident. We also had the benefit of the documents referenced.

The staff position is that the addition of a nonsafety-related containment spray system in the AP600 design would achieve an appropriate balance between prevention and mitigation of severe accidents. The staff stated that such a system would compensate for the uncertainties associated with natural removal mechanisms for aerosols during severe accidents and provide for accident mitigation and operator intervention capability as part of a long-term accident management strategy. The staff believes that a containment spray system or equivalent is consistent with the AP600 passive design philosophy and the Commission's defense-in-depth philosophy.

The Westinghouse position is that the AP600 design meets existing regulatory prevention and mitigation criteria, including the Safety Goals. This may well be the case; however, we have not yet completed our review. Westinghouse also contends that a requirement for additional systems is neither justified nor warranted. The information presented to us by Westinghouse did not address the relevant uncertainties associated with the AP600 probabilistic risk assessment.

Ideally, the determination of the need for a containment spray system should be based on a judgment as to the levels of

uncertainties associated with aerosol depletion and overall risk, as well as on the value of additional accident management capability. The first question of interest is, what are the nature and extent of the uncertainties of concern. If all uncertainties were quantifiable, it would be fairly straightforward to determine whether sufficient defense-in-depth is built into the system by assessing the risk status with respect to the subsidiary Safety Goals (core damage frequency and large, early release frequency). At present, however, a large component of uncertainties remain unquantified. The identification of these uncertainties and the qualitative judgments regarding their impact on regulatory decisions would make the debate more specific and would enhance communication among the stakeholders.

In judging the usefulness of a containment spray system in compensating for these uncertainties, both positive and negative impacts of this system should be evaluated in a quantitative and qualitative way. A judgment based on such an evaluation would help make the decision more acceptable to stakeholders because the basis for the decision would be explicit and transparent. Furthermore, such an evaluation process would be a good first step towards the integration of risk and traditional concepts such as defense-in-depth.

Although we prefer to have the information from the evaluation outlined above, based on our current state of knowledge, we support the staff's contention that the addition of a severe accident mitigation system is appropriate. The addition of a spray system to the AP600 containment would significantly increase its effectiveness in fission product control and provide the ability to intervene and control the course of an accident. We believe, however, that the spray design concept suggested by the staff is marginally adequate.

The debate associated with this issue and the difficulty of making a decision highlight our belief that the NRC needs to develop a new policy statement that would provide more guidance on the extent and nature of defense-in-depth expected by the Commission.

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

Sincerely,

A handwritten signature in dark ink, appearing to read "R. L. Seale". The signature is fluid and cursive, with the first letters of the first and last names being capitalized and prominent.

R. L. Seale
Chairman

References:

1. ACRS letter dated June 15, 1995, from T. S. Kress, Chairman, ACRS, to James M. Taylor, Executive Director for Operations, NRC, Subject: Proposed Commission Paper on Staff Positions on Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design.
2. ACRS report dated August 15, 1996, from T. S. Kress, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: SECY-96-128, "Policy and Key Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design."
3. Memorandum dated November 12, 1996, from James M. Taylor, Executive Director for Operations, NRC, to the NRC Commissioners, Subject: Clarification of Staff Position in SECY-96-128, "Policy and Key Technical Issues Pertaining to the Westinghouse AP600 Standard Pressurized Reactor Design."
4. Memorandum dated January 15, 1997, from John C. Hoyle, Secretary, NRC, to Hugh L. Thompson, Jr., Acting Executive Director for Operations, NRC, and Karen D. Cyr, General Counsel, NRC, Subject: Staff Requirements - SECY-96-128 - Policy and Key Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design.
5. Memorandum dated February 19, 1997, for the Commissioners, from Hugh L. Thompson, Jr., Acting Executive Director for Operations, NRC, Subject: SECY-97-044, "Policy and Key Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design."
6. Memorandum dated March 18, 1997, from L. Joseph Callan, Executive Director for Operations, NRC, to Chairman Jackson, Subject: Use of Non-Safety-Related Equipment to Address Safety Concerns on Nuclear Power Plants.
7. Letter dated March 13, 1997, from Brian A. McIntyre, Westinghouse Electric Corporation, to John Hoyle, Secretary, NRC, Subject: Westinghouse Comments on SECY-97-044, "Policy and Key Technical Issues Pertaining to the Westinghouse AP600 Standard Pressurized Reactor Design."
8. Memorandum dated May 16, 1997, from L. Joseph Callan, Executive Director for Operations, NRC, to the NRC Commissioners, Subject: Westinghouse Comments on SECY-97-044, "Policy and Key Technical Issues Pertaining to the Westinghouse AP600 Standard Pressurized Reactor Design."

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

December 12, 1997

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

Dear Chairman Jackson:

SUBJECT: CREDIT FOR CONTAINMENT OVERPRESSURE TO PROVIDE ASSURANCE OF
SUFFICIENT NET POSITIVE SUCTION HEAD FOR EMERGENCY CORE COOLING AND
CONTAINMENT HEAT REMOVAL PUMPS

We initially discussed this subject during the 442nd meeting of the Advisory Committee on Reactor Safeguards in June 1997. After that meeting, and an exchange of letters with the EDO, we requested additional discussions with the NRC staff to address some of our continuing concerns. We addressed these concerns during our Plant Operations Subcommittee meeting on December 2, 1997, and during the 447th meeting of the ACRS on December 3-6, 1997. We also had the benefit of the documents referenced.

Recommendation

In our June 17, 1997, letter to the Executive Director for Operations regarding our review of Generic Letter (GL) 97-04, "Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps," we stated that giving credit for containment overpressure to provide assurance of sufficient net positive suction head (NPSH) should not be considered an acceptable corrective action for any deficiencies found by the reviews requested by GL 97-04. As a result of further review of this issue, we now concur with the NRC staff position that selectively granting credit for small amounts of overpressure for a few cases may be justified. We recommend that instead of using qualitative arguments and restricting attention to a limited range of accident sequences, the decisionmaking process should consider the time variation of NPSH for a broad range of accident sequences such as typically found in a probabilistic risk assessment (PRA).

Discussion

We recognize that previous NRC staff actions have established a precedent for giving credit for containment overpressure to assure sufficient NPSH. These actions have been based on the judgment that the consequences of reduced margins are small. These actions, however, depart from the basic guidance found in Regulatory Guide (RG) 1.1, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps," and the associated Standard Review Plan Section 6.2.2, "Containment Heat Removal Systems." RG 1.1 states that "emergency core cooling and containment heat removal systems should be designed so that adequate NPSH is provided to system pumps, assuming maximum expected temperatures of pumped fluids and no increase in containment pressure from that present before any postulated LOCA." The language in Section 6.2.2 of the Standard Review Plan backs away slightly from this guidance by stating that the NPSH analysis should be based on the assumption that the containment pressure equals the vapor pressure of the sump water.

The guidance in RG 1.1 was developed from a design-basis approach that utilizes surrogate design-basis accidents (DBAs) with conservative assumptions so that the resulting design is robust (has acceptable risk) for all accident sequences. Ideally, the justification for any departures from such guidance should be grounded in the proposed Regulatory Guide 1.174 (formerly DG-1061), "An Approach for Using PRA in Risk-Informed Decisions on Plant-Specific Changes to the Current Licensing Basis," risk-informed process. In the absence of such justification, we believe that the original RG 1.1 guidance should be the appropriate regulatory position. Because of the nature of the DBA approach, however, it is difficult to quantify the impact on risk caused by the minor departures from this guidance associated with granting credit for containment overpressure to assure adequate NPSH. Therefore, the staff has relied on engineering judgment and qualitative arguments.

The potential for loss of adequate NPSH results from overheating of the sump water. The qualitative arguments used by the staff involved evaluation of selected accident sequences to show that when the sump is hot, overpressure will exist, and when overpressure is not available, the sump temperature will be limited to approximately 212° F. We conclude that these arguments do demonstrate an acceptable qualitative approach to assuring that the risk is small. We are concerned, however, with the completeness of the staff's evaluations with respect to the full spectrum of accident sequences. We recommend that future decisions

be guided by a more extensive PRA evaluation of the NPSH status for the specific plant of interest over a broader range of accident sequences, rather than relying completely on the above qualitative arguments.

The margins in NPSH afforded by the DBA approach constitute a level of defense in depth. Allowing more credit for containment overpressure reduces defense in depth. The staff's justification for this was that the consequences of losing NPSH would not be catastrophic, i.e., the particular pumps at issue would not suffer damage and the discharge flow rates would remain sufficiently high. We believe that the evidence to support these assertions needs to be identified as a part of the decisionmaking process.

In general, where departures from DBA requirements are being considered, we continue to believe that such departures should be justified using the risk-informed approach described in the proposed Regulatory Guide 1.174 (formerly DG-1061).

Sincerely,

R. L. Seale
Chairman

References:

1. Letter dated June 17, 1997, from R. L. Seale, Chairman, ACRS, to L. Joseph Callan, Executive Director for Operations, NRC, Subject: Proposed Final Generic Letter, "Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps."
2. Letter dated August 15, 1997, from L. Joseph Callan, Executive Director for Operations, NRC, to R. L. Seale, Chairman, ACRS, Subject: Proposed Final Generic Letter, "Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps."
3. Memorandum dated September 9, 1997, from John T. Larkins, Executive Director, ACRS, to L. Joseph Callan, Executive Director for Operations, NRC, Subject: Proposed Final Generic Letter "Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps."

4. U. S. Nuclear Regulatory Commission, Generic Letter 97-04: Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps, dated October 7, 1997.
5. Memorandum dated November 24, 1997, from M. Wayne Hodges, Office of Nuclear Regulatory Research and Gary M. Holahan, Office of Nuclear Reactor Regulation, to John Larkins, Executive Director, ACRS, Subject: General Regulatory Guide (DG-1061) and Standard Review Plan (SRP-Chapter 19) for Risk Informed Regulatory Decisionmaking for Plant Specific CLB Changes.

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

March 24, 1997

MEMORANDUM TO: L. Joseph Callan
Executive Director for Operations

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

SUBJECT: DRAFT COMMISSION PAPER TO PROVIDE A STATUS REPORT
AND MODIFIED APPROACH FOR FIRE PROTECTION
RULEMAKING

During the 439th meeting of the Advisory Committee on Reactor Safeguards, March 6-8, 1997, the Committee decided not to review the subject paper. The Committee, however, plans to review the comprehensive and integrated plan for transitioning to a risk-informed, performance-based regulatory structure for fire protection requirements after it has been developed by the staff.

Reference

Memorandum dated February 14, 1997, from David L. Morrison, Director, Office of Nuclear Regulatory Research, to Frank J. Miraglia, Office of Nuclear Reactor Regulation, Subject: Review of Commission Paper to Provide a Status Report and Modified Approach for Fire Protection Rulemaking.

cc: J. Hoyle, SECY
J. Blaha, OEDO
J. Mitchell, OEDO
W. Morris, RES
G. Holahan, NRR
J. Cortez, RES

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

February 12, 1997

MEMORANDUM TO: Hugh L. Thompson, Jr.
Acting Executive Director for Operations

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

SUBJECT: PROPOSED FINAL GENERIC LETTER, "REVISED CONTENTS
OF MONTHLY OPERATING REPORT"

During the 438th meeting of the Advisory Committee on Reactor Safeguards, February 6-8, 1997, the Committee considered the subject Generic Letter. The Committee decided not to review this matter, and it has no objection to the issuance of this Generic Letter.

Reference:

Memorandum dated December 4, 1996, from Ashok C. Thadani, Office of Nuclear Reactor Regulation, to Edward L. Jordan, CRGR Chairman, Subject: Request for Final Endorsement of the Proposed Generic Letter Titled "Revised Contents of Monthly Operating Report".

cc: J. Hoyle, SECY
J. Blaha, OEDO
J. Mitchell, OEDO
D. Ross, AEOD
T. Martin, NRR
J. Cortez, RES

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

February 12, 1997

MEMORANDUM TO: Hugh L. Thompson, Jr.
Acting Executive Director for Operations
FROM: John T. Larkins, Executive Director
Advisory Committee on Reactor Safeguards
SUBJECT: PROPOSED SUPPLEMENT TO GENERIC LETTER 95-06,
"CHANGES IN THE OPERATOR LICENSING PROGRAM"

During the 438th meeting of the Advisory Committee on Reactor Safeguards, February 6-8, 1997, the Committee decided not to review the subject proposed supplement.

Reference:

Memorandum dated December 23, 1996, from Bruce A. Boger, Office of Nuclear Reactor Regulation, to Edward L. Jordan, Chairman, Committee to Review Generic Requirements, Subject: Proposed Supplement to Generic Letter 95-06, "Changes in the Operator Licensing Program"

cc: J. Hoyle, SECY
J. Blaha, OEDO
J. Mitchell, OEDO
B. Boger, NRR
S. Richards, NRR
F. Guenther, NRR
J. Cortez, RES

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

March 14, 1997

MEMORANDUM TO: L. Joseph Callan
Executive Director for Operations

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

SUBJECT: PROPOSED GENERIC LETTER REGARDING LOSS OF REACTOR
COOLANT INVENTORY AND ASSOCIATED POTENTIAL FOR LOSS
OF EMERGENCY MITIGATION FUNCTIONS WHILE IN A
SHUTDOWN CONDITION

During the 439th meeting of the Advisory Committee on Reactor Safeguards, March 6-8, 1997, the Committee decided to review the subject proposed final Generic Letter after reconciliation of public comments.

Reference

Memorandum dated February 10, 1997, from T. Martin, NRR, to D. Meyer, ADM, Subject: Notice of Opportunity for Public Comment for a Proposed Generic Communication Regarding Loss of Reactor Coolant Inventory and Associated Potential for Loss of Emergency Mitigation Functions While in a Shutdown Condition (M92635).

cc: J. Hoyle, SECY
J. Blaha, OEDO
J. Mitchell, OEDO
T. Martin, NRR
A. Chaffee, NRR
J. Cortez, RES

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

June 17, 1997

Mr. L. Joseph Callan
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Mr. Callan:

SUBJECT: PROPOSED FINAL GENERIC LETTER, "ASSURANCE OF SUFFICIENT
NET POSITIVE SUCTION HEAD FOR EMERGENCY CORE COOLING AND
CONTAINMENT HEAT REMOVAL PUMPS"

During the 442nd meeting of the Advisory Committee on Reactor Safeguards, June 11-14, 1997, we reviewed the subject Generic Letter. During this review, we had the benefit of discussions with representatives of the NRC staff and of the document referenced.

It is clear that the issue of net positive suction head assurance is of high risk significance. There is, therefore, some urgency to identifying the status of the affected plants and to rectifying any noncompliances found. Thus, we support the issuance of this Generic Letter. We, however, offer the following comments.

In some cases, corrective actions will have to be taken to ensure sufficient net positive suction head. The staff needs to define acceptable corrective actions. We believe that allowing some level of containment overpressure is not an acceptable corrective action because adequate overpressure may not be present when needed. In particular, it may not be available during shutdown and containment bypass accidents.

Information obtained by the NRC inspection program is essential in providing assurance that the licensing basis is maintained. We note that most of the instances of noncompliance associated with the subject Generic Letter had remained undetected for many years. We plan to follow-up on this finding to assess how the inspection process can be made more effective.

Sincerely,

A handwritten signature in dark ink, appearing to read "R. L. Seale", is written over the typed name.

R. L. Seale
Chairman

Reference:

Memorandum to D. F. Ross, Chairman, Committee To Review Generic Requirements, from F. J. Miraglia, NRR, transmitting proposed final Generic Letter, "Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps," provided to the ACRS on May 15, 1997.

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

June 17, 1997

MEMORANDUM TO: L. Joseph Callan
Executive Director for Operations

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

SUBJECT: RESOLUTION OF THE MULTIPLE SYSTEM RESPONSES
PROGRAM ISSUES

During the 442nd meeting of the Advisory Committee on Reactor Safeguards, June 11-14, 1997, the Committee discussed your May 29, 1997 response to the ACRS letter dated June 3, 1996, to James M. Taylor, then EDO, regarding resolution of the Multiple System Responses Program (MSRP) Issues. The Committee remains interested in the full resolution of the MSRP issues and looks forward to a briefing on the staff's summary report when it is completed.

Reference:

Letter dated May 29, 1997, from L. Joseph Callan, Executive Director for Operations, to Robert L. Seale, Chairman, ACRS, Subject: Resolution of the Multiple System Responses Program Issues

cc: J. Hoyle, SECY
J. Blaha, OEDO
J. Mitchell, OEDO
L. Shao, RES
A. Thadani, RES
J. Cortez, RES

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

June 18, 1997

Mr. L. Joseph Callan
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Mr. Callan:

SUBJECT: PROPOSED GENERIC LETTER, "POTENTIAL FOR DEGRADATION OF THE EMERGENCY CORE COOLING SYSTEM AND THE CONTAINMENT SPRAY SYSTEM AFTER A LOSS-OF-COOLANT ACCIDENT BECAUSE OF CONSTRUCTION AND PROTECTIVE COATING DEFICIENCIES AND FOREIGN MATERIAL IN THE CONTAINMENT"

During the 442nd meeting of the Advisory Committee on Reactor Safeguards, June 11-14, 1997, we met with representatives of the NRC staff to discuss the subject Generic Letter. We also had the benefit of the document referenced.

Conclusions and Recommendations

- The information requested in this Generic Letter should enable the staff to determine whether licensees' use of protective coatings inside reactor containments is in compliance with the current licensing basis.
- As a result of several generic communications on the subject of loss of Emergency Core Cooling System and Containment Spray System functions caused by foreign material inside the containment, licensees have put in place programs and procedures to deal with this problem. Because these generic communications have failed to rectify this problem, it is essential that the NRC increase its efforts to ensure that the industry brings this issue to closure.
- Recent failures of qualified coatings at two PWRs call into question the validity of licensee analyses that assume qualified coatings are effective over the lifetime of the plant and will survive in a loss-of-coolant-accident environment. Additional research is needed to determine whether there are generic problems with the qualified coating application or the qualification process.

Mr. L. Joseph Callan

- 2 -

- The failure to detect foreign materials, which, in some cases, have been present since construction, raises questions about the effectiveness of inspection programs.
- Adequate tools are not available for inspectors to evaluate the quality of protective coatings. This may be a suitable topic for NRC-sponsored research.

We wish to have an opportunity to review the proposed final version of this Generic Letter after the staff has reconciled the public comments.

Sincerely,

R. L. Seale
Chairman

Reference:

1. Memorandum dated May 7, 1997, from Marylee M. Slosson, Office of Nuclear Reactor Regulation, to David L. Meyer, Office of Administration, transmitting Proposed Generic Letter, Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System After a Loss-of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in the Containment (TAC No. M97146)

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

September 9, 1997

MEMORANDUM TO: L. Joseph Callan,
Executive Director for Operations

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

SUBJECT: PROPOSED FINAL GENERIC LETTER "ASSURANCE OF
SUFFICIENT NET POSITIVE SUCTION HEAD FOR
EMERGENCY CORE COOLING AND CONTAINMENT HEAT
REMOVAL PUMPS"

During the 444th meeting of the Advisory Committee on Reactor Safeguards, the ACRS discussed your August 15, 1997 response to the Committee's comments on the subject generic letter.

The Committee remains concerned with regard to allowing credit by the staff for containment overpressure to ensure sufficient net positive suction head, and would like to continue discussion of this matter with the NRC staff during a future meeting.

The Committee also accepts your offer to be briefed on the NRC power reactor inspection program.

Reference:

Memorandum dated August 15, 1997, from L. Joseph Callan, EDO, to R. L. Seale, ACRS, Subject: Proposed Final Generic Letter "Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps"

cc: J. Hoyle, SECY
J. Blaha, OEDO
J. Mitchell, OEDO
S. Collins, NRR
F. Miraglia, NRR
S. Newberry, NRR
M. Hodges, RES
J. Cortez, RES

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

September 11, 1997

MEMORANDUM TO: L. Joseph Callan
Executive Director for Operations
FROM: John T. Larkins, Executive Director
ACRS
SUBJECT: PROPOSED GENERIC LETTER, "LABORATORY TESTING OF
NUCLEAR-GRADE ACTIVATED CHARCOAL"

During the 444th meeting of the Advisory Committee on Reactor Safeguards, September 3-5, 1997, the Committee considered the proposed Generic Letter. The Committee plans to review the proposed final Generic Letter after reconciliation of public comments.

Reference:

Memorandum dated August 27, 1997, from Frank J. Miraglia, Office of Nuclear Reactor Regulation, to Thomas T. Martin, Chairman, Committee to Review Generic Requirements, Subject: Request for Review and Endorsement of the Proposed Generic Letter Titled, "Laboratory Testing of Nuclear-Grade Activated Charcoal"

cc: J. Hoyle, SECY
J. Blaha, OEDO
J. Mitchell, OEDO
F. Miraglia, NRR
G. Holahan, NRR
M. Knapp, RES
T. Martin, AEOD
J. Cortez, RES

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

September 15, 1997

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

Dear Chairman Jackson:

SUBJECT: PROPOSED GENERIC LETTER AND DRAFT REGULATORY GUIDE DG-1074 CONCERNING STEAM GENERATOR TUBE INTEGRITY

During the 444th meeting of the Advisory Committee on Reactor Safeguards, September 3-5, 1997, we met with representatives of the NRC staff and the Nuclear Energy Institute (NEI) to discuss the proposed generic letter related to steam generator tube integrity and associated Regulatory Guide. Our Subcommittees on Materials and Metallurgy and on Severe Accidents met on August 26-27, 1997, to discuss the subject matter. We also had the benefit of the documents referenced.

The staff requests in the proposed generic letter that licensees change the technical specifications related to their steam generator tube integrity programs and reactor coolant system activity. The staff stated that the intent of the requested changes is to ensure that steam generator tube integrity is maintained consistent with the licensing basis and to provide a consistent and enforceable standard for regulating steam generator tube integrity. The draft Regulatory Guide DG-1074, "Steam Generator Tube Integrity," provides guidance for developing acceptable steam generator tube integrity programs.

Recommendation

We believe that the specifications contained in the proposed generic letter and DG-1074 will provide improvements by requiring condition monitoring (as-found condition of tubes), operational assessment (assurance of tube integrity between inspections), and nondestructive examination qualification, and will be consistent with programs that have already been implemented by many utilities. Licensee technical specifications should be amended to incorporate these requirements. We believe that the proposed generic letter and DG-1074 should be issued for public comment. We believe that

the amended technical specifications will establish regulatory consistency throughout the industry and provide enforceable performance criteria.

Discussion

The guidance proposed in DG-1074 is largely performance based and focuses on defining the characteristics of effective monitoring, assessment, and qualification programs. Such guidance should provide sufficient flexibility for the industry to introduce new inspection technologies and use updated information on steam generator tube degradation rates and mechanisms. This guidance also provides a regulatory framework for the introduction of alternate repair criteria.

Because the use of alternate repair criteria could potentially lead to increased susceptibility to thermally induced tube failures during certain severe-accident sequences, licensees wanting to introduce such criteria will be required to perform risk assessments. We support this general position, but note that the specific guidance for performing these assessments will be contained in draft Regulatory Guide DG-1073, "An Approach to Plant-Specific, Risk-Informed Decision Making: Induced Steam Generator Tube Rupture," which is being developed.

The staff has adequately addressed the issues raised during our previous meetings that are relevant to the proposed generic letter and DG-1074. We would like, however, to review the formal response to the Differing Professional Opinion, and the proposed resolution of Generic Safety Issue 163, "Multiple Steam Generator Tube Leakage," before they are issued or reviewed by the Commission. Because it is still under development, we cannot determine whether DG-1073 will adequately address the issues we have raised regarding risk assessment under severe accidents.

The Electric Power Research Institute, NEI, and PWR Owners Groups have developed industry guidelines for implementing steam generator tube integrity programs. This industry initiative is intended to strengthen existing program guidelines related to enhanced non-destructive examination, condition monitoring, and operational assessment. Since no documentation describing these guidelines has been made available to us, we cannot comment on its relation to the guidance proposed by the staff.

A representative of NEI stated that NEI is anxious to meet with the NRC staff to gain a better understanding of the staff's approach and schedule, and to resolve any differences between the NRC and industry guidelines. The NRC and NEI staffs should try to integrate the NRC and industry guidelines into a consistent and supportive framework. This effort should not delay issuance of the proposed generic letter and DG-1074 for public comment.

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

Sincerely,

R. L. Seale
Chairman

References:

1. Memorandum dated August 12, 1997, from Thomas T. Martin, Office of Nuclear Reactor Regulation, NRC, to John Larkins, Executive Director, ACRS, Subject: ACRS Review of the Proposed Steam Generator Generic Letter and Draft Regulatory Guide DG-1074, "Steam Generator Tube Integrity"
2. Report dated November 20, 1996, from T. S. Kress, Chairman, ACRS, to James M. Taylor, Executive Director for Operations, NRC, Subject: Proposed Rule on Steam Generator Integrity
3. Memorandum dated May 23, 1997, from L. Joseph Callan, Executive Director for Operations, NRC, to the Commissioners, Subject: Steam Generator Rulemaking
4. Memorandum dated June 27, 1997, from L. Joseph Callan, Executive Director for Operations, NRC, to the Commissioners, Subject: J. Hopenfeld's Differing Professional Opinion Concerning Voltage-Based Repair Criteria for Steam Generator Rulemaking
5. Report dated June 20, 1997, from R. L. Seale, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: Proposed Regulatory Approach Associated with Steam Generator Integrity
6. Letter dated July 15, 1997, from L. Joseph Callan, Executive Director for Operations, NRC, to Robert L. Seale, Chairman, ACRS, Subject: Proposed Regulatory Approach Associated with Steam Generator Integrity
7. U. S. Nuclear Regulatory Commission Draft Regulatory Guide DG-1570, "Risk Assessment of Severe Accident-Induced Steam Generator Tube Rupture," May 1997
8. U. S. Nuclear Regulatory Commission Regulatory Analysis, "Regulatory Approach for Steam Generator Tube Integrity," May 1997

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

October 15, 1997

MEMORANDUM TO: L. Joseph Callan
Executive Director for Operations

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

SUBJECT: PROPOSED FINAL GENERIC LETTER, "DEGRADATION OF STEAM
GENERATOR INTERNALS"

During the 445th meeting of the Advisory Committee on Reactor Safeguards, October 2-3, 1997, the Committee decided not to review the subject proposed final generic letter. The ACRS Subcommittee on Materials and Metallurgy discussed a draft version of this generic letter on April 15, 1997.

Reference:

Memorandum dated September 3, 1997, from Frank J. Miraglia, Office of Nuclear Reactor Regulation, to Thomas T. Martin, Chairman, Committee to Review Generic Requirements, Subject: Request for Review and Endorsement of a Proposed Generic Letter, "Degradation of Steam Generator Internals."

cc: J. Hoyle, SECY
J. Blaha, OEDO
J. Mitchell, OEDO
F. Miraglia, NRR
G. Lainas, NRR
S. Coffin, NRR
G. Wunder, NRR

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

October 15, 1997

MEMORANDUM TO: L. Joseph Callan
Executive Director for Operations

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

SUBJECT: PROPOSED FINAL GENERIC LETTER, "STEAM GENERATOR
TUBE INSPECTION TECHNIQUES"

During the 445th meeting of the Advisory Committee on Reactor Safeguards, October 2-3, 1997, the Committee decided not to review the subject proposed final generic letter. The ACRS Subcommittee on Materials and Metallurgy discussed a draft version of this generic letter on April 15, 1997.

Reference:

Memorandum dated June 23, 1997, from Frank J. Miraglia, Office of Nuclear Reactor Regulation, to Denwood F. Ross, Jr., Chairman, Committee to Review Generic Requirements, Subject: Request for CRGR Review and Endorsement of a Proposed Generic Letter Regarding Steam Generator Tube Inspection Techniques

cc: J. Hoyle, SECY
J. Blaha, OEDO
J. Mitchell, OEDO
F. Miraglia, NRR
P. Rush, NRR
A. Dromerick, NRR

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

November 13, 1997

Mr. L. Joseph Callan
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Mr. Callan:

SUBJECT: PROPOSED FINAL GENERIC LETTER, "LOSS OF REACTOR COOLANT INVENTORY AND ASSOCIATED POTENTIAL FOR LOSS OF EMERGENCY MITIGATION FUNCTIONS WHILE IN A SHUTDOWN CONDITION"

During the 446th meeting of the Advisory Committee on Reactor Safeguards, November 6-7, 1997, we reviewed the subject Generic Letter. During this review, we had the benefit of discussions with representatives of the NRC staff and the Nuclear Energy Institute. We also had the benefit of the documents referenced. During our 439th meeting, March 6-8, 1997, we decided to review this Generic Letter after the NRC staff reconciled the public comments.

The subject Generic Letter addresses an event that occurred at the Wolf Creek Generating Station, Unit 1, on September 17, 1994. We previously heard an information briefing on this event during our 419th meeting in March 1995. Wolf Creek was in a hot shutdown mode at the time of the event, and was re-borating one train of the residual heat removal (RHR) system. Concurrently, in the other RHR train, maintenance was being performed on a valve that isolates this system from the refueling water storage tank (RWST). This activity required that the valve be opened, which established a drain down path from the reactor coolant system (RCS) to the RWST that lowered the reactor vessel water level to the top of the hot leg. If this condition had been left uncorrected, the core would have been uncovered in about one hour by continued boiloff and venting of steam through an open pathway that bypassed containment. In addition, hot water from the RCS entered a pathway into a common header from which the emergency core cooling system (ECCS) pumps take suction from the RWST, which could cause steam binding of the ECCS pumps. The event had the potential to initiate an accident sequence that would include the loss of ECCS and RHR capability with bypass of containment. The staff characterized this event as the most risk-significant precursor of 1994. The conditional core damage probability was estimated at

3E-3. The staff also noted that there had been 19 previous operating events in which RCS water was inadvertently transferred to the RWST.

Results of the scoping studies performed for the staff indicate that core damage events during hot shutdown do not constitute a significant fraction of risks posed by shutdown and low-power operations. The experiential data provided by this event and by other events identified in a study performed by the NRC Office for Analysis and Evaluation of Operational Data reveal a greater probability of risk-significant events during hot shutdown than indicated by the scoping studies. We recommend that the staff reassess its understanding of risk during low-power and shutdown operations.

We believe that the Generic Letter is appropriate and should be issued promptly. Three years is too long for issuance of a Generic Letter on an event of this significance. The staff should determine the reasons for such a delay and reassess the procedures for developing and issuing generic letters.

Sincerely,

R. L. Seale
Chairman

References:

1. Memorandum dated October 21, 1997, from Tim Collins, NRR, to John T. Larkins, ACRS, transmitting: (1) Revised draft Wolf Creek Generic Letter and comparative text version identifying changes made to version issued for public comments; (2) Public comment letters received from publishing draft Generic Letter in the Federal Register; (3) Statement of Public Comment Resolution and NRC Staff Response to public comments.
2. U.S. Nuclear Regulatory Commission, NUREG/CR-4674, Volume 21, "Precursors to Potential Severe Core Damage Accidents: 1994. A Status Report," Oak Ridge National Laboratory, December 1995.
3. U. S. Nuclear Regulatory Commission, NUREG/CR-6144, Volume 1, "Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Surry, Unit 1," Summary of Results, Brookhaven National Laboratory, October 1995.

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

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

December 9, 1997

MEMORANDUM TO: L. Joseph Callan
Executive Director for Operations

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

SUBJECT: PROPOSED GENERIC LETTER, "AUGMENTED INSPECTION OF
PRESSURIZED-WATER REACTOR CLASS 1 HIGH-PRESSURE
SAFETY INJECTION PIPING"

During the 447th meeting of the Advisory Committee on Reactor Safeguards, December 3-6, 1997, the Committee decided not to review the subject proposed Generic Letter. The Committee would like the opportunity to review the document that summarizes the responses to the Generic Letter.

Reference:

Memorandum dated November 19, 1997, from Frank J. Miraglia, Office of Nuclear Reactor Regulation, to Thomas T. Martin, Chairman, Committee to Review Generic Requirements, Subject: Request for Endorsement of the Proposed Generic Letter Entitled, "Augmented Inspection of Pressurized-Water Reactor Class 1 High-Pressure Safety Injection Piping."

cc: J. Hoyle, SECY
J. Blaha, OEDO
J. Mitchell, OEDO
S. Collins, NRR
J. Strosnider, NRR
K. Wichman, NRR
M. Mitchell, NRR

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

December 10, 1997

MEMORANDUM TO: L. Joseph Callan
Executive Director for Operations

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

SUBJECT: PROPOSED GENERIC LETTER ON INTERIM GUIDANCE FOR UPDATING FINAL
SAFETY ANALYSIS REPORTS

During the 447th meeting of the Advisory Committee on Reactor Safeguards, December 3-6, 1997, the Committee considered the proposed NRC Generic Letter 97-XX, "Interim Guidance Regarding Updating Final Safety Analysis Reporting in Accordance with 10 CFR 50.71(e)." The Committee decided to continue its review of this matter following the reconciliation of public comments. The Committee looks forward to working with the staff in evaluating proposed industry guidance and in developing proposed final NRC guidance.

References:

1. Proposed NRC Generic Letter 97-XX, "Interim Guidance Regarding Updating Final Safety Analysis Reporting in Accordance with 10 CFR 50.71(e)," received November 6, 1997.
2. Letter dated November 14, 1997, from Anthony R. Pietrangelo, NEI, to Jack W. Roe, NRR, Subject: "Draft Industry Update Guidelines for Final Safety Analysis Reports."

cc: J. Hoyle, SECY
J. Blaha, OEDO
J. Mitchell, OEDO
S. Collins, NRR
J. Roe, NRR
T. Martin, AEOD
M. Knapp, RES

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

December 10, 1997

MEMORANDUM TO: L. Joseph Callan
Executive Director for Operations

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

SUBJECT: YEAR 2000 COMPUTER ISSUE

During the 447th meeting of the Advisory Committee on Reactor Safeguards, December 3-6, 1997, the Committee considered the "Year 2000 Computer Issue." The Committee may decide to review this matter following reconciliation of public comments.

References:

1. SECY-97-213, Memorandum dated September 24, 1997, from A.J. Galante, Chief Information Officer, NRC, to The Commissioners, Subject: "Year 2000 Computer Issue."
2. Letter dated November 6, 1997, from James W. Davis, NEI, to Jared S. Wermiel, NRC, Subject: "Nuclear Utility 2000 Readiness," and attached NEI/NUSMG 97-07.
3. Draft Generic Letter 97-XXX, "Year 2000 Readiness of Computer Systems at Nuclear Power Plants," received December 1, 1997.

cc: J. Hoyle, SECY
J. Blaha, OEDO
J. Mitchell, OEDO
S. Collins, NRR
B. Boger, NRR
T. Martin, AEOD
M. Knapp, RES
A. Galante, CIO

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

February 13, 1997

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

Dear Chairman Jackson:

SUBJECT: HUMAN PERFORMANCE PROGRAM PLAN

During the 438th meeting of the Advisory Committee on Reactor Safeguards, February 6-8, 1997, we completed our review of the NRC activities identified in the Human Performance Program Plan (HPPP). Our Subcommittee on Human Factors met on September 20 and December 3, 1996, to review these activities. During these reviews, we had the benefit of discussions with representatives of the staff.

In your remarks of December 2, 1996, to all NRC employees, you stated:

As we move to an era of nuclear power industry restructuring and declining NRC and industry resources, it is imperative that we are able to diagnose potentially declining licensee performance as early as possible.

We agree with your assessment. We believe that an appropriate HPPP would contribute significantly to the development of such diagnostic tools.

Conclusions and Recommendations

1. The HPPP is not a plan. It is, instead, an inventory of human performance projects within the agency. The HPPP should state explicitly what its goals are, what research efforts will be required to achieve these goals, and when and how it will be known that they have been achieved. The ownership of the present plan is diffuse. The success of such a plan as well as its dynamic nature require that ownership of the entire plan be clearly assigned.
2. A well-planned research effort in human performance is urgently needed to support both the regulation of plant operations and the transition to risk-informed and

performance-based regulation. The overall perspective that can be provided by high-level models of human performance would be helpful in the planning of this research effort. A number of such models are reviewed in NUREG/CR-6350.

3. The development of indicators of a good safety culture, the design of a meaningful human performance reporting system, and the impact of downsizing and deregulation on human performance should be major elements of the research effort.

Discussion

Operational experience has shown that human performance is a major factor in the safe operation of nuclear power plants. Understanding what can go wrong at a plant requires an integrated evaluation of both hardware and human performance; i.e., the plant must be viewed as a sociotechnical system. In particular, the term "human error," which carries the implication that the operators are to be blamed, is inaccurate in many instances and one must investigate and understand the context within which plant personnel function. This context is determined by both the design and the physical conditions of the plant, as well as by the prevailing safety culture.

The development of a plan for research on human factors is certainly not a simple task. This task would be made easier and the recommendations more convincing if the task were guided by a high-level model that identifies the important elements that influence the likelihood of unsafe human acts. Various models and taxonomies have been proposed in the literature and some are beginning to receive wide acceptance. Human performance models and error classifications that could be suitable guides for developing a research plan are being used in other projects in the Office of Nuclear Regulatory Research. The models discussed in NUREG/CR-6350, along with insights from operational experience, could serve to guide the development of an HPPP.

One specific element we would like to see addressed in the HPPP is the impact of situational assessment on compliance with procedures. Investigations of actual incidents and simulator exercises from nuclear and other industries have demonstrated the importance of what Professor James Reason of the University of Manchester calls "intended violations" (circumventions) of procedures by plant personnel. The researchers who collected data from simulator exercises point out that these were not necessarily errors; the operators simply did what they felt was the optimal response to the evolving accident. We believe there is a need to understand the reasons for such deviations and how training, procedures, and the plant safety culture could be modified to eliminate "circumventions" to the extent possible.

The present HPPP contains elements that are worth pursuing. Other elements that should be contained in the HPPP include activities to gain a better understanding of the concept of safety culture and to develop indicators of a good safety culture. The human reliability analysis research project should also be part of the HPPP. We will continue to work with the staff in developing an effective HPPP.

Sincerely,

R. L. Seale
Chairman

References:

1. Memorandum dated July 31, 1996, from Cecil Thomas, Office of Nuclear Reactor Regulation, to John Larkins, ACRS Executive Director, Subject: Forwarding Human Performance Plan Rev. 1
2. Office for Analysis and Evaluation of Operational Data Report E-95-01, "Operating Events with Inappropriate Bypass or Defeat of Engineered Safety Features," July 1995
3. U. S. Nuclear Regulatory Commission, NUREG/CR-6093, "An Analysis of Operational Experience During LP&S and a Plan for Addressing Human Reliability Issues," June 1994
4. U. S. Nuclear Regulatory Commission, NUREG/CR-6265, "Multidisciplinary Framework for Analyzing Errors of Commission and Dependencies in Human Reliability Analysis," August 1995
5. U. S. Nuclear Regulatory Commission, NUREG/CR-6350, "A Technique for Human Error Analysis (ATHEANA)," May 1996
6. Reason, J.T., Human Error, Cambridge University Press, Cambridge, United Kingdom, 1990
7. R. Montmayeul, F. Mosneron-Dupin, and M. Llory, "The Managerial Dilemma between the Prescribed Task and the Real Activity of Operators: Some Trends for Research on Human Factors," Reliability Engineering and System Safety, 45:67-73, 1994
8. U. S. Nuclear Regulatory Commission, NUREG/CR-6208, "An Empirical Investigation of Operator Performance in Cognitively Demanding Simulated Emergencies," July 1994
9. International Atomic Energy Agency, Vienna, International Nuclear Safety Advisory Group, "Safety Culture," Report 75-INSAG-4, 1991
10. NRC Chairman Shirley Ann Jackson's remarks to all NRC employees, December 2, 1996

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

October 8, 1997

Mr. L. Joseph Callan
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Mr. Callan:

SUBJECT: HUMAN PERFORMANCE AND HUMAN RELIABILITY IMPLEMENTATION
PLAN

During the 445th meeting of the Advisory Committee on Reactor Safeguards, October 2-3, 1997, we reviewed the human performance and human reliability implementation plan. Our Subcommittee on Human Factors reviewed this matter during a meeting on June 3, 1997. During these reviews, we had the benefit of discussions with representatives of the NRC staff and of the documents referenced.

Conclusions and Recommendations

1. The development of the Plan is still in progress. Although the mission of the human performance activities has been defined, it is not clear how the six programs listed in the plan will satisfy this mission. This may be due to the fact that the high-level model that we recommended in our February 13, 1997 report to show the connection between the objectives of these programs and Agency activities has not yet been developed.
2. Although ATHEANA appears to be a reasonable starting point for the development of the high-level model, it needs to be modified and expanded to serve the needs of the Plan.
3. We recommend that insights derived from operational experience be incorporated into the development of the high-level model.
4. We recommend that development of the Plan be given high priority, even if some of the activities listed in Appendix B of the Plan must be suspended.

Discussion

In our February 13, 1997 report, we commented that the Human Performance Program Plan was not a plan, but an inventory of human performance projects. Much progress has been made since then. The ATHEANA model has been put forth as a first step toward the development of a high-level model for integrating and prioritizing Agency activities associated with human performance. The staff told us that this high-level model has not yet been completed. Consequently, we cannot evaluate whether individual activities that the Agency is currently sponsoring will lead to an effective treatment of human performance.

We believe that the development of a high-level model remains urgent. It will allow the staff to articulate the objectives of the Plan and to show how individual activities contribute to the achievement of these objectives. It will also provide the basis for prioritizing these activities. Without such a high-level model, the need for many of the activities listed in Appendix B of the Plan is not apparent.

The use of ATHEANA as the starting point for the development of the high-level model is promising. The staff has acknowledged that ATHEANA, originally intended to serve as the basis for quantifying human error rates in probabilistic risk assessments, will have to be expanded and modified to serve the needs of the Plan. We agree that a model suitable for developing a research program in support of the Plan will be different from the ATHEANA model.

We emphasize the importance of incorporating insights from operational experience into the high-level model. We urge that the most significant lessons learned from operational experience be included in the current effort.

During our discussions with the staff, it became evident that the development of the Plan had been slowed, perhaps due to lack of adequate resources. We reiterate that the development of the Plan should be given high priority, even if some of the activities listed in Appendix B of the Plan must be suspended. We believe that the Plan will enable the staff to systematically prioritize and efficiently manage the Agency human performance related activities.

We will continue to work with the staff in this area.

Sincerely,

A handwritten signature in cursive script, appearing to read "R. L. Seale".

R. L. Seale
Chairman

References:

1. Memorandum dated August 12, 1997, from M. Wayne Hodges, Office of Nuclear Regulatory Research, to John T. Larkins, Executive Director, ACRS, Subject: Transmittal of Human Performance and Human Reliability Implementation Plan.
2. ACRS report dated February 13, 1997, from R.L. Seale, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: Human Performance Program Plan.

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

February 11, 1997

MEMORANDUM TO: Hugh L. Thompson Jr.
Acting Executive Director for Operations
FROM: John T. Larkins, *John T. Larkins* Executive Director
Advisory Committee on Reactor Safeguards
SUBJECT: DRAFT AEOD REPORT ON GRID PERFORMANCE FACTORS

During the 438th meeting of the Advisory Committee on Reactor Safeguards, February 6-8, 1997, the Committee decided not to review the subject report.

Reference:

U. S. Nuclear Regulatory Commission, Office for Analysis and Evaluation of Operational Data, AEOD S96-XX, "Grid Performance Factors," December 1996 (Draft)

cc: J. Hoyle, SECY
J. Blaha, OEDO
J. Mitchell, OEDO
D. Ross, AEOD
C. Rossi, AEOD
J. Rosenthal, AEOD
M. Wegner, AEOD

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

June 23, 1997

Mr. L. Joseph Callan
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Mr. Callan:

SUBJECT: REGULATORY GUIDANCE FOR IMPLEMENTATION OF DIGITAL
INSTRUMENTATION AND CONTROL SYSTEMS

During the 442nd meeting of the Advisory Committee on Reactor Safeguards, June 11-14, 1997, we reviewed the proposed final update of Standard Review Plan (SRP), Chapter 7, "Instrumentation and Controls," including Branch Technical Positions (BTPs) and associated Regulatory Guides related to digital instrumentation and control (I&C) systems. We also discussed the staff's integration of insights from the National Research Council Phase 2 Study Final Report and a draft safety evaluation report on an Electric Power Research Institute (EPRI) Topical Report, TR-106439, "Guideline on Evaluation and Acceptance of Commercial Grade Digital Equipment for Nuclear Safety Applications." During this review, we had the benefit of discussions with representatives of the NRC staff and its contractor, the Lawrence Livermore National Laboratory (LLNL). We previously met with the National Research Council panel to discuss its Phase 2 study Final Report during our 439th meeting, March 6-8, 1997. We also had the benefit of the documents referenced.

Conclusions and Recommendations

- We are satisfied that the guidance provided in the update to SRP Chapter 7 is adequate to ensure safety and therefore recommend issuance of the proposed final SRP update and associated guidance.
- We recommend approval of the draft safety evaluation report on the EPRI Topical Report, TR-106439.
- The staff should develop criteria for a graded approach to the review process of digital I&C systems based on safety significance.

- The staff should participate in workshops to discuss the application of the SRP Chapter 7 update and the use of associated regulatory guidance. The staff may also find it useful to solicit industry participation for a pilot application for I&C systems using the SRP Chapter 7 update.
- The staff should work with the industry to clarify the "system level" issue in Generic Letter 95-02, "Use of NUMARC/EPRI Report TR-102348, 'Guideline on Licensing Digital Upgrades,' in Determining the Acceptability of Performing Analog-to-Digital Replacements under 10 CFR 50.59."
- For systems that contain digital components, the effects of software failures should be included in the probabilistic risk assessment (PRA).

Discussion

We previously raised issues regarding the graded approach based on importance to safety, the level of detail provided in the Regulatory Guides, the balance in the guidance between the review of the design process and the assessment of the product, as well as the linkage between Chapter 7 and other SRP chapters.

The staff has proposed to adopt the criteria in draft Regulatory Guide DG-1064 for graded quality assurance (GQA) to determine the level of review required for digital I&C systems. We previously expressed concerns with the proposed version of DG-1064 in our report to Chairman Jackson dated March 17, 1997. We do, however, support consideration of an appropriate graded approach for the review of digital I&C systems.

Case studies presented by the staff provide evidence that a well-managed review based on the proposed regulatory guidance can assure that there is balance between the review of the design process and the review of the product. Additional research is necessary to address the specific needs for digital I&C systems and to develop more efficient, improved quantitative means of assessing software quality and performance. The research proposed by the staff is a reasonable first step.

The National Research Council Phase 2 Study Final Report included a recommendation that the staff reexamine its position on the "systems level" issue. The staff has concluded that no further action is required. We believe that the staff should work with industry to reconcile the NUMARC/EPRI guidelines and consider revising Generic Letter 95-02. The lack of clarity on this issue inhibits implementation of digital system technology in nuclear power plants.

For some time, we have raised questions about the treatment of software and digital system reliability in PRAs. The National Research Council report addresses this issue and recommends that the staff develop methods for estimating failure probabilities of software-based digital systems, including systems that use commercial, off-the-shelf software and hardware. We support the research on these topics initiated by the staff.

In summary, there will be a continuing need to maintain and update SRP Chapter 7 and associated guidance as the technology changes and becomes more widely used. Similarly, there is a need for continued effort to clarify guidance for the industry through public workshops and pilot programs.

Sincerely,

R. L. Seale
Chairman

References:

1. Memorandum dated May 7, 1997, from B. Boger, Office of Nuclear Reactor Regulation (NRR), to J. Larkins, ACRS, Subject: Proposed Final Updated SRP Chapter 7 on Instrumentation and Controls and Final Regulatory Guides on Digital Computer Software Quality, and attached SRP sections, Branch Technical Positions, and Regulatory Guides.
2. Draft Safety Evaluation Report dated May 5, 1997, from D. Matthews, NRR, to R. Torok, EPRI, Subject: Review of EPRI Topical Report TR-106439, Guideline on Evaluation and Acceptance of Commercial Grade Equipment for Nuclear Safety Applications.
3. Electric Power Research Institute, Topical Report, EPRI TR-106439, "Guideline on Evaluation and Acceptance of Commercial Grade Digital Equipment for Nuclear Safety Applications," Final Report, October 1996.
4. Generic Letter 95-02, "Use of NUMARC/EPRI Report TR-102348, 'Guideline on Licensing Digital Upgrades,' in Determining the Acceptability of Performing Analog-to-Digital Replacements under 10 CFR 50.59."
5. Committee on Application of Digital Instrumentation and Control Systems to Nuclear Power Plant Operations and Safety, Board on Energy and Environmental Systems, Commission on Engineering and Technical Systems, National Research Council, "Digital Instrumentation and Control Systems in Nuclear Power Plants," Safety and Reliability Issues - Final Report, 1997.
6. NUMARC/EPRI Topical Report TR-102348, "Guideline on Licensing Digital Upgrades," December 1993.

7. Letter dated October 23, 1996, from T. S. Kress, ACRS Chairman, to J. Taylor, Executive Director for Operations, NRC, Subject: Draft Update of Standard Review Plan, Chapter 7, Instrumentation and Controls.
8. Letter dated June 6, 1996, from T. S. Kress, ACRS Chairman, to J. Taylor, Executive Director for Operations, NRC, Subject: Regulatory Guidance Documents Related to Digital Instrumentation and Control Systems.
9. Report dated October 13, 1995, to Shirley A. Jackson, NRC Chairman, from T. S. Kress, ACRS Chairman, Subject: National Academy of Sciences/National Research Council Study on Digital Instrumentation and Control Systems in Nuclear Power Plants, Safety and Reliability Issues - Phase 1.

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

June 20, 1997

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

Dear Chairman Jackson:

SUBJECT: PROPOSED REGULATORY APPROACH ASSOCIATED WITH STEAM
GENERATOR INTEGRITY

During the 442nd meeting of the Advisory Committee on Reactor Safeguards, June 11-14, 1997, we met with representatives of the NRC staff and the Nuclear Energy Institute (NEI) to discuss the staff's proposed regulatory approach for ensuring steam generator tube integrity. We reviewed the staff's original rulemaking approach during our November 7-9, 1996 meeting, and provided a letter to the Executive Director for Operations on November 20, 1996. Our Subcommittees on Materials and Metallurgy and on Severe Accidents also reviewed the rulemaking approach on June 3-4 and November 5-6, 1996, and on January 9 and April 15-16, 1997. We also had the benefit of the documents referenced.

The staff had intended to develop a risk-informed, performance-based rule to provide a new regulatory basis for managing steam generator tube degradation. But in the course of the development of the new rule, the staff discovered that the introduction of new tube performance criteria to replace the current 40-percent through-wall crack-depth criterion could create a potential for increased susceptibility to thermally induced tube failure during certain severe accident sequences.

We previously reviewed the staff's analyses of tube performance in severe accident situations. Although there were many uncertainties in the analyses, we concurred with the staff position that thermally induced tube failures must be addressed when alternate tube repair criteria are considered.

The introduction of a requirement to evaluate tube behavior during severe accidents would constitute a backfit. The regulatory analysis performed by the staff demonstrated that this backfit could not be justified on a generic basis.

The need still exists for regulatory guidelines for addressing new diverse steam generator tube degradation mechanisms. The staff has determined that enforceable guidance, which also provides flexibility to licensees, can be promulgated with a generic letter and two regulatory guides. One regulatory guide, which would be required for licensees electing to adopt alternate tube repair criteria, would provide guidance for risk assessments that address thermally induced tube failures.

The generic letter and the other regulatory guide would specify new guidelines for inspecting and assessing the condition of degraded tubes. The intent of these new guidelines is to ensure that nondestructive examination techniques are adequate for assessing tube conditions. The staff believes that these new guidelines constitute a compliance backfit and are not subject to the backfit rule.

The NEI Steam Generator Working Group contends that the existing regulatory requirements and guidance are adequate, and that the proposed generic letter, which would request changes to licensee technical specifications, is unnecessary. NEI has presented a framework of guidelines, which address a wide variety of steam generator related issues, that could serve as the basis for industry guidance and obviate the need for the proposed regulatory guides. NEI agrees that a risk assessment is appropriate when alternate repair criteria are adopted.

Recommendation

The proposed staff approach for developing a new regulatory basis to ensure steam generator tube integrity is reasonable, and we recommend that the staff continue with this approach.

Discussion

The industry and the staff are in crisis with respect to the present regulatory requirements for steam generator tube integrity. The mechanisms by which steam generator tubes degrade are now different than those envisaged at the time applicable regulations were formulated. The industry is forced to repair steam generator tubes with defects that cannot be characterized accurately or confidently. Both the industry and staff agree that this results in plugging or repairing tubes that otherwise should function satisfactorily well into the future. Such a practice saps the economic viability of the plant and curtails the resource base available to support plant operations. At the same time, the staff is responding on a case-by-case basis to licensee requests for staff action concerning identified steam generator tube degradation. Actions are needed to define criteria for steam generator maintenance programs that will ensure steam generator

tube integrity, and to develop inspection guidelines for the NRC staff.

The three options available to provide stability to the steam generator program are rulemaking, technical specification changes initiated through the generic letter process, or reliance on industry programs. Rulemaking would provide fixed enforceable criteria but is untenable due to backfit considerations. Reliance on formal industry programs would provide flexibility to licensees but would be unenforceable through the regulatory process. The proposed changes that would be made in the technical specifications would result in enforceable criteria and inspectable solutions, and would, according to the staff, constitute a compliance backfit.

There is agreement between the staff and NEI that improvements in condition monitoring, operational assessment, and nondestructive examination qualification programs are beneficial and have been implemented by most utilities. We believe that the staff and NEI should reach agreement on the performance criteria for ensuring steam generator tube integrity, as well as on the guidelines for proper implementation of these criteria.

The staff proposal would allow licensees to continue to use older criteria with augmented condition monitoring and operational assessment. On the other hand, a licensee could institute alternate repair criteria specific to the particular degradation mechanisms peculiar to the plant's steam generators. The regulatory guidance would define how to qualify this degradation specific management program. Licensees choosing this option would have to evaluate the risk of induced steam generator tube failure during severe accidents.

We agree with the staff decision to address probabilistic risk assessment issues in a separate regulatory guide consistent with the draft Regulatory Guide DG-1061, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Current Licensing Basis."

We plan to continue our review of this issue when the implementing documents are available.

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

Sincerely,

A handwritten signature in dark ink, appearing to read "R. L. Seale". The signature is fluid and cursive, with the first letters of the first and last names being capitalized and prominent.

R. L. Seale
Chairman

References:

1. Memorandum dated May 23, 1997, from L. Joseph Callan, Executive Director for Operations, to NRC Commissioners, Subject: Steam Generator Rulemaking.
2. Letter dated November 20, 1996, from T. S. Kress, Chairman, ACRS, to James M. Taylor, Executive Director for Operations, NRC, Subject: Proposed Rule on Steam Generator Integrity.
3. U.S. Nuclear Regulatory Commission, Draft Regulatory Guide DG-1061, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Current Licensing Basis," dated February 28, 1997.

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

September 10, 1997

Mr. L. Joseph Callan
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Mr. Callan:

SUBJECT: BOILING WATER REACTOR PRESSURE VESSEL SHELL WELD
INSPECTION RECOMMENDATIONS (BWRVIP-05)

During the 444th meeting of the Advisory Committee on Reactor Safeguards, September 3-5, 1997, we reviewed the BWRVIP-05 report and the associated staff interim Safety Evaluation Report. During this review, we had the benefit of discussions with representatives of the NRC staff and of the documents referenced. Our subcommittees on Materials and Metallurgy and on Severe Accidents also reviewed this matter during a meeting on August 26, 1997.

The Commission has required in 10 CFR 50.55a(g)(6)(ii)(A)(2) that all boiling water reactor (BWR) and pressurized water reactor (PWR) vessel weld inspections comply with the requirements specified in Item B1.10 of Examination Category B-A, "Pressure Retaining Welds in Reactor Vessel," in Table IWB-2500-1 of subsection IWB of section XI, Division 1, of the ASME Boiler and Pressure Vessel Code. The requirement is for 100-percent inspection of all welds. The requirement for 100-percent inspection represents the culmination of a consistent, conservative shift that has occurred over the years in the ASME Code requirements. Licensees with BWRs are requesting that the requirement be modified to allow inspections of only axial welds in the vessel.

The reactor vessel is a fundamentally important barrier to prevent the release of radionuclides from a nuclear power plant. It is essential that there be very high confidence in the integrity of vessel welds to have confidence in the plant safety. Compliance with the requirements for 100-percent inspection of all reactor vessel welds can be viewed as an element of the defense-in-depth regulatory philosophy to ensure protection of the public.

The NRC staff and industry agree that welds in BWR vessels are, on the whole, less vulnerable to inservice failure than welds in PWR vessels. The PWR vessels, however, have, in general, received 100-percent inspection of all welds in accordance with the code

requirements and thus have an additional degree of assurance against the existence of gross defects. The lower vulnerability of the BWR vessel welds is due primarily to the lower probability of events that pose threats to the welds and to the lower neutron fluences to the vessel walls.

Reactor vessel welds were inspected during fabrication using radiography, magnetic particle, and dye penetrant methods, which are effective for detecting flaws that break the metal surface and volumetric defects. These methods are insensitive for detecting subsurface, crack-like flaws. During fabrication, some welds were even inspected using ultrasonic techniques of the day that are inferior to techniques now available.

The prudence of inservice vessel weld inspections in BWRs does not appear to be a contentious issue. Both the staff and industry agree that it is necessary to ensure against inservice degradation such as stress corrosion cracking and to mitigate concerns that undocumented repairs, which could introduce defects, may have been made to the welds. The contentious issue is how complete must be the inspection and whether anything less than 100-percent inspection of all welds will erode defense-in-depth. The staff and industry calculations show that the failure probabilities of reactor vessels due to fabrication flaws in circumferential welds are very low ($< 10^{-8}/\text{yr}$). Inspection lowers these failure probabilities by about a factor of three. Since the failure probabilities are already so low, there is little contribution to defense-in-depth from inspection of these welds.

Industry argues that inspections should focus on the axial welds, since these pose the most risk. The proposed 100-percent inspection of the axial welds, which constitute about 40-percent of the total welds, provides assurance against inservice degradation. The results of the analyses performed by the staff, although not directly comparable to those done by the industry, support the conclusion that circumferential welds are orders of magnitude less vulnerable to inservice failures than axial welds.

The industry proposal to inspect less than 100-percent of the vessel welds is essentially a request for a change in the current licensing basis for BWRs as a class. This request should be handled using the risk-informed process now being developed by the NRC staff. The staff appears to be proceeding in accordance with the guidance in proposed Regulatory Guide DG-1061, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Current Licensing Basis," subject to the constraint that the defense-in-depth regulatory philosophy is preserved. The analyses performed to date by the staff and industry appear to suggest that the relaxed inspection requirements produce minimal changes in risk.

Some additional efforts are needed, however, to address uncertainties associated with these analyses, such as showing that flaw size distributions input to the models are justifiable and consistent with available data, including those obtained in past inspections of the welds.

Data from inspections of welds in PWR vessels may also be of use in the definition of realistic flaw distributions. The adequacy of the Marshall flaw size distribution, which is used by both the staff and industry in probabilistic fracture mechanics calculations, needs careful consideration. This distribution predicts that the fraction of flaws larger than a specific size decreases exponentially with flaw size. It is this exponential decrease that is chiefly responsible for the very large differences in failure probability between axial and circumferential welds. It is not clear that the experts who formulated this distribution as a bound on the frequency of large flaws intended for the distribution to be used to compare the relative frequencies of the approximately 2 cm flaws that lead to failure in axial welds and the approximately 4 cm flaws that lead to failure in circumferential welds. Indeed, models like RR-PRODIGAL suggest that the frequency of large flaws does not decrease exponentially with flaw size, although the departure from the exponential behavior may occur only for flaw sizes greater than those responsible for the large differences in failure probabilities between axial and circumferential welds.

The uncertainty in the nature of the flaw distribution is the most critical factor in determining the relative probability of failure between circumferential and axial welds. Additional uncertainties need to be addressed to more accurately assess the actual failure probabilities of BWR vessel welds. To address such uncertainties, a comprehensive analysis is needed of accident sequences that pose threats to the welds. Careful consideration should be given to operator actions that affect the probability of challenges to the integrity of BWR vessel welds. The uncertainty analysis should address conservatisms inherent in existing calculations, including the assumptions that weld flaws penetrate through the cladding and that all flaws are located on the inner surface of the vessel.

The staff should consider the relative value of partial inspections of the welds. In truth, 100-percent inspection of welds will seldom be practicable because equipment configurations in BWRs will limit access to some welds. Limitations in the capability to detect flaws by inspections need to be recognized as well.

It may not be possible to make risk-informed decisions concerning inspections of BWR vessel welds as a class. Decisions will probably have to be made on a plant-specific basis. The probability of vessel failure by loss of weld integrity will be dependent on irradiation-induced embrittlement of the vessel and

the frequency of challenges to weld integrity. In this regard, we encourage the staff to continue development of a database on vessel embrittlement specific to BWRs.

The staff and industry analyses done to date consider only the 40-year licensing life. These analyses show that conditional risks of vessel failure increase rapidly in the later years of plant life. Indeed, the failure probabilities at the end of 40 years of plant life are predicted for some plants to be surprisingly high, approaching or surpassing bounding vulnerabilities assumed for plant probabilistic risk assessments. There are a number of conservatisms in the staff and industry analyses, which were primarily intended to compare the relative probability of failure of axial and circumferential welds. The apparent rapid increase of vessel failure probability at 40 years, however, suggests that it is important to also analyze failure probabilities for plant life extensions beyond the current license limit. In these analyses, the contributions of base metal failure should be included.

We plan to review the staff's proposed Final Safety Evaluation Report and related matters.

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

Sincerely,

R. L. Seale
Chairman

References:

1. Letter dated September 28, 1995, from J. T. Beckham, Jr., Southern Nuclear Operating Company, to U. S. Nuclear Regulatory Commission, Document Control Desk, Subject: BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations
2. Letter dated August 14, 1997, from Brian W. Sheron, Office of Nuclear Reactor Regulation, NRC, to Carl Terry, Niagara Mohawk Power Company, Subject: Transmittal of NRC Staff's Independent Assessment of the Boiling Water Reactor Vessel and Internals Project BWRVIP-05 Report and Proprietary Request for Additional Information
3. SECY-97-088, Memorandum dated April 22, 1997, from L. Joseph Callan, Executive Director for Operations, NRC, for the Commissioners, Subject: NRC Staff's Position on Augmented Examination Requirements for Boiling Water Reactor Pressure Vessels Pursuant to 10 CFR 50.55a(g)(6)(ii)(A)
4. U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor

Mr. L. Joseph Callan

- 5 -

Regulation, NRC Information Notice 97-63: Status of NRC
Staff's Review of BWRVIP-05, August 7, 1997

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

October 10, 1997

Mr. L. Joseph Callan
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Mr. Callan:

**SUBJECT: RESOLUTION OF THE DIFFERING PROFESSIONAL OPINION RELATED TO
STEAM GENERATOR TUBE INTEGRITY**

During the 445th meeting of the Advisory Committee on Reactor Safeguards, October 2-3, 1997, we reviewed the proposed resolution of the subject differing professional opinion (DPO). On several previous occasions, we have reviewed the issues raised in the DPO and related matters. During these reviews, we had the benefit of discussions with representatives of the NRC staff and the author of the DPO. We also had the benefit of the documents referenced.

The DPO initially arose during the development of Generic Letter (GL) 95-05 "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside-Diameter Stress-Corrosion Cracking." The staff recognizes that the use of alternate repair criteria such as the voltage-based limit for outside-diameter stress corrosion cracking (ODSCC) in tube support plates might permit operations with the presence of short through-wall cracks. The issues raised in the DPO focus on concerns that arise when steam generators are permitted to operate with such degradation. The concerns raised in the DPO bear upon GL 97-xx, "Steam Generator Tube Integrity," and the draft regulatory guides DG-1074, "Steam Generator Tube Integrity," and DG-1073, "An Approach for Plant-Specific, Risk-Informed Decision Making; Induced Steam Generator Tube Rupture," only insofar as to whether they provide adequate guidance for the development of alternate repair criteria.

We believe that the DPO-issues-response document prepared by the staff accurately summarizes the issues associated with the DPO. The staff grouped the issues into five related areas and noted they are not independent. These areas are: nondestructive examination (NDE) reliability; leakage under design-basis accident (DBA) conditions, including main steamline break (MSLB); increased frequency of core damage with containment bypass; nonconservative estimates of iodine spiking during depressurization transients, such as an MSLB; and increases in the frequency of thermally induced tube failures under certain severe-accident scenarios.

In his recent presentation to us, the DPO author agreed that two of these issues, iodine spiking and thermally induced tube failure, had been resolved adequately by the staff. He felt that the remaining three issues noted below were properly stated by the staff, but he did not concur with the proposed resolution of these issues: NDE reliability; leakage under DBA conditions including MSLB; the increased frequency of core damage with containment bypass. The last two issues are closely linked.

The higher frequency of core damage with containment bypass estimated by the author of the DPO is largely due to the much higher levels of leakage that he projects to occur during DBAs. The DPO-issues-response document notes that leakage through degraded tubes under DBAs was addressed in GL 95-05 for voltage-based repair criteria for ODSCC at tube support plates. In GL 95-05, an empirical correlation relating leakage to eddy-current voltage is used to estimate leakage under DBAs. Recognizing that such empirical correlations involve large uncertainties, the staff proposed that these uncertainties be addressed through performance monitoring and statistical analyses, and the staff provided guidelines for such analyses and monitoring.

The author of the DPO argues that such an approach does not adequately assess the uncertainties in the leakage estimates. As discussed in draft NUREG-1477, "Voltage-Based Interim Plugging Criteria for Steam Generator Tubes," such empirical correlations are in much better agreement with existing laboratory data and field experience on leakage than estimates based on the fracture mechanics cited by the author of the DPO.

We believe the DPO-issues-response document provides a good statement of the issues and represents a serious attempt by the staff to resolve them. We believe that the staff should issue this document for public comment.

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

Sincerely,

R. L. Seale
Chairman

References:

1. Memorandum dated May 23, 1997, from L. Joseph Callan, Executive Director for Operations, NRC, to NRC Commissioners, Subject: Steam Generator Rulemaking.
2. Letter dated November 20, 1996, from T. S. Kress, Chairman, ACRS, to James M. Taylor, Executive Director for Operations, NRC, Subject: Proposed Rule on Steam Generator Integrity.
3. Memorandum dated June 27, 1997, from L. Joseph Callan, Executive Director for Operations, NRC, to NRC Commissioners, Subject: J. Hopenfeld's Differing Professional Opinion Concerning Voltage-Based Repair Criteria for Steam Generator Tubes: Steam Generator Rulemaking.
4. Memorandum dated September 19, 1997, from Brian W. Sheron, Office of Nuclear Reactor Regulation, NRC, to John Larkins, ACRS, Subject: ACRS Review of the Differing Professional Opinion Related to Steam Generator Tube Integrity (Predecisional).
5. Memorandum dated August 12, 1997, from T. T. Martin, Office of Nuclear Reactor Regulation, NRC, to John Larkins, ACRS, Subject: ACRS Review of the Proposed Steam Generator Generic Letter (Predecisional).
6. U. S. Nuclear Regulatory Commission, Draft Regulatory Guide DG-1073, "An Approach for Plant-Specific, Risk-Informed Decision Making: Induced Steam Generator Tube Rupture," Revised August 26, 1997,

7. U. S. Nuclear Regulatory Commission, NUREG-1477, "Voltage-Based Interim Plugging Criteria for Steam Generator Tubes," June 1993.
8. U. S. Nuclear Regulatory Commission, NRC Generic Letter 95-05: Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking, August 3, 1995

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

April 18, 1997

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

Dear Chairman Jackson:

SUBJECT: ESTABLISHING A BENCHMARK ON RISK DURING LOW-POWER AND
SHUTDOWN OPERATIONS

This report is to draw attention to the critical need for developing an understanding of risk posed by low-power and shutdown operations at nuclear power plants. This need is apparent as a result of: (1) repeated events during these modes of power plant operations, (2) changes being made in plant operations in response to economic forces, and (3) the ongoing NRC initiative to develop risk-informed, performance-based regulation. We believe it is essential that the NRC staff undertake a quantitative examination of risk during low-power and shutdown operations at representative nuclear power plants. That is, the NRC staff needs to establish a high-quality benchmark on risk during low-power and shutdown operations comparable to that which it has derived for risk during power operations from the NUREG-1150 study [Ref. 1] and other sources. The benchmark for risk during low-power and shutdown operations should address the following:

- a representative range of plant types,
- all phases of low-power and shutdown operations,
- accidents initiated by internal fires and other external events,
- human performance, the unusual source term, radionuclide dispersal, and on-site populations that will affect the predictions of accident consequences, and
- uncertainties to a depth similar to that done for the risk benchmark for power operations.

A substantial effort will be required to develop the technical capabilities to conduct this benchmark risk analysis. Results of the benchmark risk analysis may suggest the need for refinements to the Commission's Safety Goals. In particular, the Commission may find from the results that it wants to specify limits on the tolerable durations of plant configurations that pose very high risks.

Our recommendation for a detailed benchmark analysis is based on the results of scoping risk studies done by the staff contractors [Refs. 2,3], the continuing string of worrisome events at plants during low-power and shutdown operations, and assessments of the risk significance of plant events by the Office for Analysis and Evaluation of Operational Data. The staff's contractors have done limited analyses of risk during one phase of shutdown operations at a pressurized water reactor with a subatmospheric containment [Ref. 2] and one phase of shutdown operations at a Mark III boiling water reactor [Ref. 3]. Results of these studies show that even when the risk for a short period of shutdown operations is normalized over a full calendar year, the risk is a significant fraction of the risk calculated for the same plant during power operations:

Boiling Water Reactor

	Power Operations	Shutdown Operations*
Mean Core Damage Frequency	4.1×10^{-6}	2.1×10^{-6}
Mean Early Fatality Risk	8.2×10^{-9}	1.4×10^{-8}
Mean Latent Cancer Fatality Risk	9.5×10^{-4}	3.8×10^{-3}

* Plant Operating Mode 5 (cold shutdown) only.

Pressurized Water Reactor

	Power Operations	Shutdown Operations*
Mean Core Damage Frequency	4.1×10^{-5}	4.2×10^{-6}
Mean Early Fatality Risk	2.0×10^{-6}	4.9×10^{-8}
Mean Latent Cancer Fatality Risk	5.2×10^{-3}	1.6×10^{-2}

* Mid-loop operation only.

These partial results, however, may not adequately reflect current operating practices. The industry has instituted new guidelines [Ref. 4] for low-power and shutdown operations that are intended to reduce risk. Several licensees are using software such as the Electric Power Research Institute's ORAM (Outage Risk Assessment

and Management) to plan activities during low-power and shutdown operations. These software tools are based on risk insights derived from simplified probabilistic risk assessments. If the NRC staff is to provide effective safety oversight of low-power and shutdown operations, the staff will have to understand the technical bases of the software tools and the approximations in risk assessments that have been used to develop these tools. The availability of benchmark risk assessments for low-power and shutdown operations for representative plants akin to the benchmark risk assessments for power operations appears to be essential for the development of this understanding.

Despite the new guidelines and software tools developed by the nuclear power industry for low-power and shutdown operations, events that reveal safety vulnerabilities of the plants continue to occur. Among the more recent of these events are:

- The Wolf Creek plant was in a "hot shutdown" condition when activities involving the residual heat removal system created a flow path that allowed approximately 9,200 gallons of reactor coolant to transfer to the refueling water storage tank. Had this draining not been promptly terminated, the operability of the emergency core cooling system would have been compromised. The Accident Sequence Precursor Analysis indicated that this event had a high conditional core damage probability [Ref. 5]. The scoping studies of shutdown risk, however, suggested that a "hot shutdown" condition was of such a low risk significance that it did not merit quantification.
- Loss of core cooling was threatened by the formation of a nitrogen bubble in the reactor coolant system at the Haddam Neck plant as a result of an improper valve lineup. Injection of high-pressure nitrogen into the reactor vessel continued for over three days while the plant was in a "cold shutdown" condition. The water level in the reactor vessel was believed to have been displaced three feet below the vessel flange [Ref. 6].
- At the Cooper plant, about 10,000 gallons of water was inadvertently lost from the refueling cavity because a submerged valve was opened to the main steam line drains. It took over an hour for operators to identify the source of the loss of coolant inventory [Ref. 7].

Events during low-power and shutdown operations are consuming significant staff resources. At our meeting on December 5, 1996, we were told that more than 50 percent of recent events requiring Augmented Inspection Teams have occurred when plants were in low-power or shutdown conditions. Human errors during these conditions appear to be especially probable. A number of incidents that have

occurred during low-power and shutdown conditions are reviewed in the report NUREG/CR-6093 [Ref. 8]. This report concluded that factors influencing operator actions are different from those typically regarded as important during full-power operations and states that: "Unlike full-power operations, large numbers of multiple concurrent tasks are possible during LP&S (low-power and shutdown) conditions. This has implications for both the PRA (probabilistic risk assessment) modeling process and the HRA (human reliability assessment) quantification process."

We are concerned that this situation will be exacerbated as the industry moves to longer cycle times with less frequent opportunities to exercise its low-power and shutdown operating procedures. The situation may also be exacerbated by industry efforts to shorten the duration of low-power and shutdown operations by increasing the intensity of activities during these periods. The industry will want to relieve burdens during outages by doing some maintenance while the plant is operating. For the staff to approve a trade-off between maintenance "on-line" and maintenance during outages, it will have to consider risk. To do this, the staff will have to gain an understanding of risk during low-power and shutdown operations commensurate with its understanding of risk during power operations.

The staff is now embarked on an effort to develop risk-acceptance criteria for providing regulatory relief to licensees. Staff judgments on these matters are based on a firm foundation concerning event probabilities during power operations derived both from the Individual Plant Examinations done by licensees and from its own benchmarking risk studies reported in NUREG-1150. There is no comparable basis for making judgments concerning the accident probabilities and risk during low-power and shutdown operations. At present, there is no defensible regulatory basis to determine the extent to which results obtained for power operations ought to be augmented to account for risk of low-power and shutdown operations. A more complete understanding of risks during all phases of nuclear plant operations is essential to ensure that regulations address real, significant risks and do not impose ad hoc measures to correct discovered deficiencies in the hope that these measures will also address risk-significant issues.

We believe it is essential for the success of the Commission's effort to adopt risk-informed, performance-based regulation that a more complete understanding of the full spectrum of risk be established on a defensible technical basis. This more complete understanding is needed now as pivotal decisions are being made on the implementation of risk-informed, performance-based regulation. We do not believe that existing scoping analyses or further scoping efforts will establish adequate benchmarks concerning risk during low-power and shutdown operations. This is especially so in light

of evidence that time-dependent human performance is important. Significant efforts may be needed to establish new risk assessment methods and to understand phenomena associated with core damage events and the dispersal of radioactivity during these phases of plant operations. We are confident that areas of substantial uncertainty will arise in the assessment of risk during low-power and shutdown operations. Defensible quantification of these uncertainties will require the same type of effort that was needed to quantify uncertainties in risk during power operations.

It will take time to develop a usefully complete understanding of risk during low-power and shutdown operations. We recommend that a well-planned, deliberate effort with realistic time schedules and extensive peer review be undertaken first to develop methods and technologies that may be needed and then to benchmark risk during low-power and shutdown operations.

Sincerely yours,

R. L. Seale
Chairman

References:

1. U.S. Nuclear Regulatory Commission, NUREG-1150, Vol. 1, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," December 1990.
2. U.S. Nuclear Regulatory Commission, NUREG/CR-6144, BNL-NUREG-52399, Vol. 1, Brookhaven National Laboratory, "Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Surry, Unit 1," Summary of Results, October 1995.
3. U.S. Nuclear Regulatory Commission, NUREG/CR-6143, SAND93-2440, Vol.1, Sandia National Laboratories, "Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Grand Gulf, Unit 1," Summary of Results, July 1995.
4. NUMARC 91-06, Nuclear Management and Resources Council, Inc., "Guidelines for Industry Actions to Assess Shutdown Management," December 1991.
5. U.S. Nuclear Regulatory Commission, Information Notice 95-03, Supplement 1, "Loss of Reactor Coolant Inventory and Potential Loss of Emergency Mitigation Functions While in a Shutdown Condition," dated March 25, 1996.
6. U.S. Nuclear Regulatory Commission, Information Notice 94-36, Supplement 1, "Undetected Accumulation of Gas in Reactor Coolant System," November 1996.

7. U.S. Nuclear Regulatory Commission, AEOD/S96-02, "Assessment of Spent Fuel Pool Cooling," September 1996.
8. U.S. Nuclear Regulatory Commission, NUREG/CR-6093, "An Analysis of Operational Experience During Low Power and Shutdown and a Plan for Addressing Human Reliability Assessment Issues," June 1994.

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

May 7, 1997

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

Dear Chairman Jackson:

SUBJECT: DESIGN BASIS VERIFICATION

During the 441st meeting of the Advisory Committee on Reactor Safeguards, May 1-3, 1997, we met with representatives of the NRC staff and the Nuclear Energy Institute to discuss design basis verification. We reviewed the staff criteria for evaluating licensee responses to the NRC request for design basis information pursuant to 10 CFR 50.54(f), staff initiatives to perform design inspections, and related industry activities and initiatives. We also had the benefit of the documents referenced.

Conclusions and Recommendations

- We believe that the current four-phase approach is effective in identifying those licensees that need to take action to maintain their design basis. The staff review of licensee responses appears to have been successful in identifying and prioritizing follow-up inspection activities.
- The design inspections conducted to date have shown that, in some cases, the actual plant configuration and procedures do not correspond to the design basis upon which the plant was licensed. This illustrates the value of the design inspection program and suggests that such a program be continued in some form.
- The NRC will be relying on the results of probabilistic risk assessments (PRAs) to an ever-increasing extent as it embarks on risk-informed, performance-based regulation. PRAs should be based on the current configuration of the plant. Results of the design basis inspections should, then, be shared formally with the Office of Nuclear Regulatory Research and the Office for Analysis and Evaluation of Operational Data. Where inconsistencies between PRA assumptions and design inspection results are found, it may be of use to conduct

sensitivity studies to establish the risk significance of these inconsistencies.

Sincerely,

R. L. Seale
Chairman

References:

1. Section (f) of 10 CFR 50.54, "Conditions of Licenses."
2. Memorandum dated March 19, 1997, from L. J. Callan, Executive Director for Operations, NRC, to the Commissioners, Subject: Update on 10 CFR 50.54(f) Response Review Efforts: Pilot Process Results.
3. Memorandum dated February 25, 1997, from L. J. Callan, Executive Director for Operations, NRC, to the Commissioners, Subject: Review of Licensees' Responses to the 10 CFR 50.54(f) Letter of October 9, 1996, on the Adequacy and Accuracy of Design Bases Information for Nuclear Power Plants.
4. Nuclear Energy Institute, NEI 96-05, draft Revision D, "Guidelines for Assessing Programs for Maintaining the Licensing Basis," dated July 25, 1996.
5. Nuclear Management and Resources Council, Inc., NUMARC 90-12, "Design Basis Program Guidelines," October 1990.

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

June 19, 1997

MEMORANDUM TO: L. Joseph Callan
Executive Director for Operations

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

SUBJECT: ESTABLISHING A BENCHMARK ON RISK DURING LOW-POWER
AND SHUTDOWN OPERATIONS

During the 442nd meeting of the Advisory Committee on Reactor Safeguards, June 11-14, 1997, the Committee discussed your June 6, 1997 response to the ACRS report dated April 18, 1997, to Chairman Jackson regarding establishing a benchmark on risk during low-power and shutdown operations. The Committee looks forward to working with the staff on this issue and would like to discuss the staff's goals, objectives, strategy, and schedules for completing this activity during a future meeting.

Reference:

Letter dated June 6, 1997, from L. Joseph Callan, EDO, to R. L. Seale, Chairman, ACRS, Subject: Establishing a Benchmark on Risk During Low-Power and Shutdown Operations.

cc: J. Hoyle, SECY
J. Blaha, OEDO
J. Mitchell, OEDO
S. Collins, NRR
A. Thadani, RES
D. Ross, AEOD
J. Cortez, RES

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

September 11, 1997

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

Dear Chairman Jackson:

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

During the 444th meeting of the Advisory Committee on Reactor Safeguards, September 3-5, 1997, we reviewed the voluntary approach proposed by the industry for reporting reliability and availability information for risk-significant systems and equipment to the NRC. Our Subcommittees on Probabilistic Risk Assessment, Plant Operations, and Fire Protection also met with representatives of the NRC staff and the industry to discuss this matter on August 29, 1997. We also had the benefit of the documents referenced.

We provided a report to the Commission on April 12, 1995, recommending publication of the proposed rule associated with this matter for public comment. In that report, we noted that high-quality, plant-specific reliability and availability information are needed if risk-based regulation is to fully reach its potential for both improving safety and reducing burdens on licensees. We also agreed with the staff position that availability and reliability data needed to support risk-based regulation should be publicly available. In a letter dated May 12, 1995, the EDO agreed with our comments and recommendations. It may also be noted that a publicly available database would help to make risk-informed decisions more understandable to the public.

The staff agrees that the voluntary approach proposed by the industry to provide reliability and availability information in a proprietary database is acceptable. The NRC has procedures for making proprietary data available to affected parties in specific cases. The proprietary database will preclude, or at least limit, independent exploration of reliability data for the evaluation of risk information. Data used by the NRC for generic assessments must also be publicly available. The staff proposes to alleviate the difficulties of the public availability of data by assembling a database that does not identify individual plants. We support

this proposal. We would like to review the utility of this publicly available database once there has been some experience with it.

ACRS Chairman Dr. Robert L. Seale did not participate in Committee's deliberations regarding this matter.

Sincerely,

R. L. Seale
Chairman

References:

1. Memorandum dated June 12, 1997, from John C. Hoyle, Secretary, NRC, to L. Joseph Callan, Executive Director for Operations, NRC, Subject: Staff Requirements - SECY-97-101, Proposed Rule, 10 CFR Section 50.76, "Reporting Reliability and Availability Information for Risk-Significant Systems and Equipment."
2. SECY-97-101, Memorandum dated May 7, 1997, from L. Joseph Callan, Executive Director for Operations, NRC, to the Commissioners, Subject: Proposed Rule, 10 CFR Section 50.76, "Reporting Reliability and Availability Information for Risk-Significant Systems and Equipment."
3. Report dated April 12, 1995, from T. S. Kress, Chairman, NRC, to Ivan Selin, Chairman, NRC, Subject: Proposed Rulemaking on Reporting Reliability and Availability Information for Risk-Significant Systems and Equipment.
4. Letter dated May 12, 1995, from James M. Taylor, Executive Director for Operations, NRC, to T. S. Kress, Chairman, ACRS, Subject: Proposed Rulemaking on Reporting Reliability and Availability Information for Risk-Significant Systems and Equipment.

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

December 16, 1997

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

Dear Chairman Jackson

SUBJECT TREATMENT OF UNCERTAINTIES VERSUS POINT VALUES
 IN THE PRA-RELATED DECISIONMAKING PROCESS

During the 443rd, 444th, 446th, and 447th meetings of the Advisory Committee on Reactor Safeguards, July 9-11, September 3-5, November 6-7, and December 3-6, 1997, respectively, we met with representatives of the NRC staff to discuss issues included in the Staff Requirements Memorandum dated May 27, 1997, regarding the use of uncertainty versus point values in the PRA-related decisionmaking process (Reference 1). Our Subcommittee on Reliability and Probabilistic Risk Assessment (RPRA) met with the staff and industry representatives to discuss these matters on July 7, August 28, October 21-22, and November 12-13, 1997.

Background

Uncertainty has always been of concern to nuclear power regulators. As early as 1956, Willard F. Libby, Acting Chairman of the Atomic Energy Commission (AEC), wrote to the Congressional Joint Committee on Atomic Energy that "it is incumbent upon the new industry and the Government to make every effort to recognize every possible event or series of events which could result in the release of unsafe amounts of radioactive material to the surroundings and to take all steps necessary to reduce to a reasonable minimum the probability that such events will occur in a manner causing serious overexposure to the public." (Reference 2)

Even though Dr. Libby used the word "probability," about 20 years would pass before systematic calculations of probabilities would be produced for the "possible event or series of events" to which he referred. The "reasonable minimum" of the unquantified probability that was achieved at that time was attained through the development and application of the concepts of defense in depth and safety margins.

Defense in depth is advocated in numerous documents as the principal means of controlling the (still unquantified) probability of accidents. For example, during the 1971 hearings on emergency core cooling, the AEC staff stated "The safety goal, therefore, is the prevention of exposure of people to this radioactivity. This goal can be achieved with a high degree of assurance, *although not perfectly* [emphasis added], by use of the concept of defense in depth. The three separate lines of the defense in depth provided for power reactors are considered appropriate to reduce to an acceptable value the probability and potential consequences of radioactive releases." (Reference 3)

Although the approaches of defense in depth and safety margins have served the industry well from the safety perspective, they were intended to be conservative and, as implemented today, they impose a heavy regulatory burden. The level of safety was not quantified. The first call for a more rational approach to regulation based on improved understanding of risk came in 1967 from F. Reginald Farmer (Reference 4) of the United Kingdom Atomic Energy Authority. The Reactor Safety Study (WASH-1400) (Reference 5) soon followed in 1975. Not surprisingly, the WASH-1400 study itself proved to be conservative in some areas, e.g., the analysis of the containment, and nonconservative in others, e.g., the analysis of earthquakes and fires. There has been tremendous progress in our understanding of the risks from nuclear power plants since that study (a history of PRA developments since WASH-1400 is given in Reference 6).

Realizing that the availability of risk numbers made it possible to reexamine the question of how safe is safe enough, the Commission issued the safety goal policy in 1986 (Reference 7). The recognition that uncertainties had to be dealt with is reflected in the following three statements from the policy statement.

Statement I "It is the Commission's intent that the risks from all the various initiating mechanisms be taken into account to the best of the capability of current evaluation techniques."

Statement II "To the extent practicable, the Commission intends to ensure that the quantitative techniques used for regulatory decisionmaking take into account the potential uncertainties that exist so that an estimate can be made on the confidence level to be ascribed to the quantitative results."

Statement III "The Commission has adopted the use of mean estimates for the purposes of implementing the quantitative objectives of this safety goal policy."

The Commission's safety goals were derived from societal considerations, i.e., independent of the PRA state of the art. Even though they were expressed both

qualitatively and quantitatively, it was clear that the Commission did not intend to simply compare a PRA "point estimate" (however it was defined) with the numerical goals

The Issue

As noted above, the numerical estimates that PRAs produce have been scrutinized to an extraordinary degree since the early days of WASH-1400. Sometimes the debate regarding the accuracy of these numbers detracts from the intended use of PRA.

It is not the intent to regulate on the basis of risk estimates alone (thus, "risk-informed" regulation). The objective is to gain enough confidence in the numerical probabilities of a set of accident scenarios so that the traditional approaches (defense in depth and safety margins) that have already been applied to this set can be better managed. This means either relaxing some existing requirements, if proven burdensome and non-contributing to risk reduction, or adding new requirements, if the traditional approaches have not covered some detrimental events.

The preceding discussion suggests that the question regarding the quality of PRA results ought not to be an absolute one, but, rather, a comparative one. Therefore, we offer the following observation:

Observation 1

When PRA results and insights are proposed to be used in the regulatory process, the question to be asked should be: To what degree is there confidence that the use of PRA results and insights will improve on the existing regulatory system for the problem of interest?

The words "PRA results and insights" include the set of dominant scenarios to risk (or core damage, as the case may be), as well as an assessment of the uncertainties regarding the frequencies of these scenarios. The utilization of PRA results and insights depends on our confidence that their use will improve the regulations in accordance with the Commission's vision. It is definitely not a case of PRA versus the traditional approach.

In Observation 1, the key words are "will improve." There is improvement when the regulations contribute to the safe and efficient use of nuclear materials, as per the recently articulated vision of the Commission: "In implementation of its mission, Nuclear Regulatory Commission actions enable the Nation to safely and efficiently use nuclear materials." (Reference 8)

Uncertainties

As our brief historical review has demonstrated, the uncertainties regarding off-normal events and incidents in nuclear power plants have been of concern since the early days of reactor regulation. In the early seventies, quantifying the uncertainties was synonymous with developing probability distributions for the failure rates and the frequencies of accident initiators. This explicit quantification of uncertainties posed a new problem to safety analysts. They soon discovered that the interpretation of the concept of probability was controversial among mathematicians. Several schools of thought were available, of which the frequentist and the Bayesian schools were dominant. When the nuclear debate was heating up in the mid-seventies, the analysts were reluctant to get involved in an additional controversy.

This attitude, although understandable in the context of the times, was unfortunate, because it led to confusion and the perception that uncertainty analysis was controversial and to be avoided. It also led to some circumlocutions. For example, the WASH-1400 treatment of failure rates is purely Bayesian, yet that voluminous report does not acknowledge this fact explicitly. Similarly, the NUREG-1150 studies (Reference 9) claimed to elicit "weighting factors" from the experts, rather than admit that they were eliciting probabilities. Although "officially," both frequentist and Bayesian viewpoints were equally valid, no PRA had been done using frequentist methods because it cannot be done. Industry-sponsored PRAs, however, have readily acknowledged using Bayesian methods in an explicit way (Reference 10).

It is now known that uncertainties in failure rates and other parameters appearing in PRA can be quantified via probability distributions using available generic and plant-specific data and appropriate Bayesian methods. The propagation of these distributions through the PRA logic diagrams is straightforward using standard computer packages. We believe that there is no excuse for failing to do an uncertainty analysis on the parameters of the PRA models. Therefore, we offer the following observation.

Observation 2

The Bayesian interpretation of probability provides the appropriate framework for PRA. Probability distributions for the parameters of PRA models, e.g., failure rates, should be developed using all available evidence and propagated to produce the probability distribution of the quantity of interest, e.g., core damage frequency (CDF) and large, early release frequency (LERF).

Since regulators must confront uncertainties, it is evident that, if PRA is to be used as in our Observation 1, the probability distributions of Observation 2 must be derived. Anything less does not represent what is actually known about these failure rates. This brings up the issue of "point estimates," for which we offer the following observation.

Observation 3.

The only "point estimates" that are unambiguously defined are those that are summary measures of a probability distribution, e.g., the mean value, the median value, and various percentile values.

Ill-defined "point estimates," such as "best estimates," have limited utility. Point estimates are valuable for screening purposes after a convincing case has been made that the uncertainties have been handled appropriately, e.g., they are either negligible or have been bounded. In fact, the use of such point values is an important tool in screening the thousands of minimal cut sets that a PRA produces. Such use, however, should be followed by a rigorous uncertainty analysis of the dominant sequences.

The uncertainties of interest in reactor regulation have been termed "state-of-knowledge" uncertainties (Reference 11) or, more recently, "epistemic" uncertainties (References 12, 13). The parameter uncertainties that are referred to in Observation 2 are only a part of the total epistemic uncertainties. Uncertainties resulting from model assumptions and approximations are also epistemic and more difficult to quantify. Examples would include models used for evaluating severe accident phenomena in Level II PRAs.

Model uncertainty is the key to any use of PRA results. When events or processes are modeled poorly or not at all, there is uncertainty that has not been quantified, in the sense that it is not part of the probability distributions produced by propagating parameter uncertainties. The fact that uncertainty is not quantified does not mean, however, that nothing is known about it. The PRA structure provides a good framework within which these uncertainties can be assessed qualitatively through sensitivity analyses or other means (see, for example, Reference 14). These uncertainties exist independently of whether or not they are quantified in PRAs. Recalling Observation 1, use of PRA insights must include a qualitative description of unquantified uncertainties, in addition to those that have been quantified. Any PRA-based argument for easing the regulatory requirements of the traditional approach is weakened when the unquantified uncertainties are very large and pertinent to the application. Therefore, we offer the following observation.

Observation 4.

Regulatory decisions must be made in the light of all the relevant uncertainties. These include the uncertainties quantified in PRAs, as well as significant unquantified uncertainties. Although "point" values, defined as in Observation 3, can be useful for screening purposes, they are summary measures of the probability distributions and should not be the sole basis for decisionmaking.

The deliberation on uncertainties that we are recommending is best accomplished by considering the scenarios that dominate the event of interest. The set of dominant scenarios is one of the most important results of PRA and has been proven to be very useful in risk management (Reference 15). A discussion of the overall uncertainties without a discussion of the sources of uncertainties is of limited value. Thus, we offer the following observation:

Observation 5.

The dominant scenarios should be an integral part of the deliberation on uncertainties.

The regulatory decisions of immediate interest are those related to requests for changes in the current licensing basis (CLB). In discussing uncertainties, it is important to consider possible benefits of the proposed change. For example, a change that reduces the regulatory burden in certain areas could allow the reallocation of resources to more risk significant issues and activities. Therefore, we offer the following observation:

Observation 6.

The unquantified uncertainties associated with a proposed change in the CLB should include the possible beneficial impact of the proposed change on plant safety.

Finally, we note that the decisionmaking process described in Regulatory Guide 1.174 treats uncertainties and point values in a manner consistent with our

recommendations as discussed in our report dated December 11, 1997 (Reference 16)

Sincerely,

R L Seale
Chairman

References

- 1 Staff Requirements Memorandum dated May 27, 1997, from John C Hoyle, Secretary of the Commission, to John T Larkins, ACRS, Subject Meeting with Advisory Committee on Reactor Safeguards on Friday, May 2, 1997
- 2 Letter dated March 14, 1956, from Willard F Libby, Acting Chairman of the Atomic Energy Commission, to Senator B Hickenlooper, Joint Committee on Atomic Energy, reproduced in D Okrent, *Nuclear Reactor Safety*, The University of Wisconsin Press, Madison, 1981
- 3 Testimony of the AEC Regulatory Staff at a Public Rulemaking Hearing on Interim Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Reactors on January 27, 1972, Issued December 28, 1971, U S Atomic Energy Commission
- 4 F Reginald Farmer, "Reactor Safety and Siting A Proposed Risk Criterion," *Nuclear Safety*, Vol 8, No 6, pp 539-548, Nov -Dec 1967
- 5 U S Nuclear Regulatory Commission, NUREG-75/014, "Reactor Safety Study, An Assessment of Accident Risks in U S Nuclear Power Plants, WASH-1400," October 1975
- 6 Eric S Beckjord, Mark C Cunningham, and Joseph A Murphy, "Probabilistic Safety Assessment Development in the United States 1972-1990," *Reliability Engineering and System Safety*, Vol 39, pp 159-170, 1993
- 7 U S Nuclear Regulatory Commission, "Safety Goals for the Operation of Nuclear Power Plants Policy Statement," *Federal Register*, Vol 51, p 30028, August 21, 1986
- 8 U S Nuclear Regulatory Commission, "Strategic Plan Fiscal Year 1997 - Fiscal Year 2000," September 1997

- 9 U S Nuclear Regulatory Commission, NUREG-1150, *"Severe Accident Risks An Assessment for Five US Nuclear Power Plants,"* December 1990
10. Pickard, Lowe, and Garrick, Inc , Westinghouse Electric Corporation, and Fauske & Associates, Inc , *"Zion Probabilistic Safety Study,"* Report prepared for Commonwealth Edison Company, Chicago, 1981
11. Stanley Kaplan and B John Garrick, "On the Quantitative Definition of Risk," *Risk Analysis*, Vol 1, No 1, pp 11-28, March 1981
- 12 George E Apostolakis, "A Commentary on Model Uncertainty," in U S Nuclear Regulatory Commission, NUREG/CP-0138, *Proceedings of Workshop I in Advanced Topics in Risk and Reliability Analysis Model Uncertainty, Its Characterization and Quantification*, October 20-22, 1993
- 13 Gareth W Parry, "The Characterization of Uncertainty in Probabilistic Risk Assessments of Complex Systems," *Reliability Engineering and System Safety*, Vol 54, pp 119-126, 1996
- 14 Dennis Bley, Stanley Kaplan, and David Johnson, "The Strengths and Limitations of PSA Where We Stand," *Reliability Engineering and System Safety*, Vol 38, pp 3-26, 1992
- 15 Mardyros Kazarians, Nathan Siu, and George Apostolakis, "Risk Management Application of Fire Risk Analysis," *Proceedings of the First International Symposium on Fire Safety Science*, pp. 1029-1038, C E Grant and P J Pagni, Editors, Hemisphere Publishing Corporation, New York, 1986
- 16 Report dated December 11, 1997, from R L Seale, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject "Proposed Final Regulatory Guide 1.174 and Standard Review Plan Chapter 19 for Risk-Informed Performance-Based Regulation"

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

February 12, 1997

MEMORANDUM TO: Hugh L. Thompson, Jr.
Acting Executive Director for Operations

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

SUBJECT: PROPOSED REVISION 3 TO REGULATORY GUIDE 1.134,
"MEDICAL EVALUATION OF LICENSED PERSONNEL AT
NUCLEAR POWER PLANTS"

During the 438th meeting of the Advisory Committee on Reactor Safeguards, February 6-8, 1997, the Committee decided not to review the proposed revision 3 to Regulatory Guide 1.134.

Reference:

Memorandum dated December 11, 1996, from Ashok C. Thadani, Office of Nuclear Reactor Regulation, to Thomas S. Kress, ACRS Chairman; K. D. Cyr, Office of the General Counsel; Edward L. Jordan, CRGR Chairman, Subject: DG-1068, Proposed Revision 3 to Regulatory Guide 1.134

cc: J. Hoyle, SECY
J. Blaha, OEDO
J. Mitchell, OEDO
A. Thadani, NRR
B. Boger, NRR
S. Richards, NRR
J. Cortez, RES

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

March 19, 1997

MEMORANDUM TO: L. Joseph Callan
Executive Director for Operations

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

SUBJECT: DRAFT REGULATORY GUIDE DG-1060, "FASB STANDARDS
FOR DECOMMISSIONING COST ACCOUNTING"

During the 439th meeting of the Advisory Committee on Reactor Safeguards, March 6-8, 1997, the Committee decided not to review the subject draft Regulatory Guide.

Reference

Memorandum dated February 12, 1997, from Bill M. Morris, Office of Nuclear Regulatory Research, to Thomas S. Kress, ACRS, and Denwood F. Ross, Office of Nuclear Regulatory Research, Subject: Draft Regulatory Guide DG-1060, "FASB Standards for Decommissioning Cost Accounting."

cc: J. Hoyle, SECY
J. Blaha, OEDO
J. Mitchell, OEDO
B. Morris, RES
B. Richter, RES
J. Cortez, RES

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

March 19, 1997

MEMORANDUM TO: L. Joseph Callan
Executive Director for Operations

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

SUBJECT: DRAFT REGULATORY GUIDE DG-1067, "DECOMMISSIONING
OF NUCLEAR POWER REACTORS"

*R. Savco
for*

During the 439th meeting of the Advisory Committee on Reactor Safeguards, March 6-8, 1997, the Committee decided not to review the subject draft Regulatory Guide.

Reference

Draft Regulatory Guide DG-1067, "Decommissioning of Nuclear Power Reactors," February 1997.

cc: J. Hoyle, SECY
J. Blaha, OEDO
J. Mitchell, OEDO
S. Weiss, NRR
M. Masnik, NRR
J. Cortez, RES

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

March 25, 1997

MEMORANDUM TO: L. Joseph Callan
Executive Director for Operations

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

SUBJECT: DRAFT REGULATORY GUIDE DG-1070, "REQUIREMENTS
FOR SAMPLING FOR DEDICATING SIMPLE, METALLIC
COMMERCIAL GRADE ITEMS IN NUCLEAR POWER
PLANTS"

During the 439th meeting of the Advisory Committee on Reactor Safeguards, March 6-8, 1997, the Committee decided to consider reviewing the subject draft regulatory guide after reconciliation of public comments. The Committee has no objection to the issuance of this draft regulatory guide for public comment.

Reference:

Draft Regulatory Guide DG-1070, "Requirements for Sampling for Dedicating Simple, Metallic Commercial Grade Items in Nuclear Power Plants," Revision 1, February 10, 1997.

cc: J. Hoyle, SECY
J. Blaha, OEDO
J. Mitchell, OEDO
L. Campbell, NRR
R. Johnson, RES
W. Gleaves, RES
J. Cortez, RES

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

July 14, 1997

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

Dear Chairman Jackson:

SUBJECT: PROPOSED REGULATORY GUIDE AND STANDARD REVIEW PLAN
CHAPTER FOR RISK-INFORMED, PERFORMANCE-BASED INSERVICE
INSPECTION

During the 443rd meeting of the Advisory Committee on Reactor Safeguards, July 9-11, 1997, we met with representatives of the NRC staff to review the proposed Regulatory Guide DG-1063 and Standard Review Plan (SRP) Chapter 3.9.8 for risk-informed, performance-based inservice inspection. Our Subcommittee on Probabilistic Risk Assessment also met on July 8, 1997 with the staff, industry representatives, and other interested parties to discuss these documents and industry initiatives. We also had the benefit of the documents referenced.

We believe that the approach described in proposed SRP Chapter 3.9.8 and Regulatory Guide DG-1063 will lead to substantial improvements in inservice inspection for piping. In response to our comments, the staff identified changes it plans to make to these documents before they are issued for public comment. We recommend that these documents be issued for public comment subject to incorporation of those changes. The staff also proposed a list of questions regarding issues that arose during our meetings, which it plans to include in the Federal Register notice to solicit public comments. We agree with these questions.

Dr. Dana Powers did not participate in the Committee's deliberations regarding draft Regulatory Guide DG-1063.

Sincerely,

R. L. Seale
Chairman

References:

Memorandum dated June 3, 1997, from B. Sheron, NRR, M. W. Hodges, RES, L. C. Shao, RES, G. Holahan, NRR to J. Larkins, ACRS, Subject: Transmittal of Pre-decisional Draft Regulatory Guide DG-1063: "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Inservice Inspection of Pipes," and Draft Standard Review Plan Chapter 3.9.8, "Standard Review Plan for the Review of Risk-Informed Inservice Inspection Applications."

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

July 15, 1997

MEMORANDUM TO: L. Joseph Callan
Executive Director *for Operations*
FROM: John T. Larkins, *John T. Larkins* Executive Director
ACRS/ACNW
SUBJECT: PROPOSED FINAL REVISION 3 TO REGULATORY GUIDE 5.44,
"PERIMETER INTRUSION ALARM SYSTEMS"

During the 443rd meeting of the Advisory Committee on Safeguards, July 9-11, 1997, the Committee decided not to review the subject regulatory guide.

Reference:

Memorandum dated June 12, 1997, from Theodore S. Sherr, NMSS, to John T. Larkins, ACRS, Subject: Regulatory Guide 5.44, "Perimeter Intrusion Alarm Systems, Rev. 3."

cc: J. Hoyle, SECY
J. Blaha, OEDO
J. Mitchell, OEDO
C. Paperiello, NMSS
T. Sherr, NMSS
P. Dwyer, NMSS
F. Miraglia, NRR
J. Cortez, RES

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

September 9, 1997

MEMORANDUM TO: L. Joseph Callan
Executive Director ~~for Operations~~
FROM: John T. Larkins, *John T. Larkins* Executive Director
ACRS
SUBJECT: PROPOSED REVISION TO REGULATORY GUIDE 5.62,
"REPORTING OF SAFEGUARDS EVENTS"

During the 444th meeting of the Advisory Committee on Reactor Safeguards, September 3-5, 1997, the Committee decided not to review the subject regulatory guide.

Reference:

Memorandum dated July 1, 1997, from Frank J. Miraglia, Office of Nuclear Reactor Regulation, to Denwood F. Ross, Jr., Chairman, Committee to Review Generic Requirements, Subject: Draft Revision to Regulatory Guide 5.62, "Reporting of Safeguards Events"

cc: J. Hoyle, SECY
J. Blaha, OEDO
J. Mitchell, OEDO
B. Miller, NMSS
F. Miraglia, NRR
L. Cunningham, NRR
R. Rosano, NRR
J. Cortez, RES

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

September 9, 1997

MEMORANDUM TO: L. Joseph Callan
Executive Director for Operations

FROM: John T. Larkins, *John T. Larkins* Executive Director
ACRS

SUBJECT: PROPOSED REVISION TO REGULATORY GUIDE 8.7,
"INSTRUCTIONS FOR RECORDING AND REPORTING
OCCUPATIONAL RADIATION EXPOSURE DATA"

During the 444th meeting of the Advisory Committee on
Reactor Safeguards, September 3-5, 1997, the Committee decided not
to review the subject regulatory guide.

Reference:

Memorandum dated July 30, 1997, from Joseph A. Murphy, Office of
Nuclear Regulatory Research, NRC, to John T. Larkins, Executive
Director, ACRS, Subject: Draft Regulatory Guide DG-8023, Proposed
Revision to Regulatory Guide 8.7, "Instructions for Recording and
Reporting Occupational Radiation Exposure Data"

cc: J. Hoyle, SECY
J. Blaha, OEDO
J. Mitchell, OEDO
F. Miraglia, NRR
J. Murphy, RES
C. Trottier, RES
M. Thomas, RES
J. Cortez, RES

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

September 10, 1997

Mr. L. Joseph Callan
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Mr. Callan:

SUBJECT: DRAFT REGULATORY GUIDE DG-1069, FIRE PROTECTION PROGRAM
FOR PERMANENTLY SHUTDOWN AND DECOMMISSIONING NUCLEAR
POWER PLANTS

During the 444th meeting of the Advisory Committee on Reactor Safeguards, September 3-5, 1997, we reviewed the subject draft Regulatory Guide. During this review, we had the benefit of discussions with representatives of the NRC staff and of the document referenced.

Draft Regulatory Guide DG-1069 is intended to provide guidance to licensees for maintaining fire protection programs for nuclear plants that are permanently shut down and defueled, and are in the process of being decommissioned. Such guidance has not heretofore been available. The risk posed to the public by permanently shutdown nuclear plants is expected to be much reduced because of lower inventories of releasable radionuclides and lower energy densities available for the breach of barriers. On the other hand, barriers to the release of radionuclides in the shutdown plants are less numerous and less robust. Also, the frequency of fire and the availability of combustibles may actually increase.

There is not now a quantitative understanding of risks posed by permanently shutdown nuclear plants. In the absence of detailed risk information, the staff has elected to formulate performance-based recommendations to licensees for satisfying the regulatory requirements for appropriate fire protection programs. The proposed Regulatory Guide endorses approximately 30 industry fire protection standards. The staff believes that implementation of these industry standards coupled with the performance of a fire hazard analysis meeting minimal requirements will ensure that the level of fire risk at permanently shutdown nuclear plants will be at or below the level of fire risk existing at operating plants. Demonstration of the equivalence of protection is apparently beyond the current state-of-the-art.

Industry standards specified in the Regulatory Guide provide explicit guidance concerning many features of fire protection programs. The Regulatory Guide is less explicit with respect to the fire hazard analysis that will be acceptable to the staff. We believe that the staff should more definitively describe its expectations for a fire hazard analysis, especially for permanently shutdown nuclear plants co-located with existing reactors and shutdown reactors with fuel still present in the spent fuel pool. This should not delay issuance of the Regulatory Guide for public comment.

The risk of fires at permanently shutdown nuclear plants will vary with time. The staff expects that licensees will be able to use the process specified in 10 CFR 50.59 to change fire protection programs as vulnerabilities decrease. The staff should ensure that any revisions to 10 CFR 50.59 or associated regulatory guides would continue to allow the licensees to modify fire protection programs.

We have no objection to the issuance of the Draft Regulatory Guide DG-1069 for public comment, and would like to have an opportunity to review the proposed final Regulatory Guide after reconciliation of public comments.

Sincerely,

A handwritten signature in dark ink, appearing to read "R. L. Seale". The signature is fluid and cursive, with the first letters of the first and last names being capitalized and prominent.

R. L. Seale
Chairman

Reference:

U. S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Draft Regulatory Guide, DG-1069, Fire Protection Program for Permanently Shutdown and Decommissioning Nuclear Power Plants, June 1997.

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

November 14, 1997

MEMORANDUM TO: L. Joseph Callan
Executive Director for Operations

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

SUBJECT: DRAFT REGULATORY GUIDE DG-1029, REVISION 1.7,
"GUIDELINES FOR EVALUATING ELECTROMAGNETIC AND
RADIO-FREQUENCY INTERFERENCE IN SAFETY-RELATED
I&C SYSTEMS"

*R. Savin
for*

During the 446th meeting of the Advisory Committee on Reactor Safeguards, November 6-7, 1997, the Committee considered the subject Regulatory Guide. As recommended by the staff, the Committee plans to review the proposed final version of this Regulatory Guide after reconciliation of public comments.

Reference:

Memorandum dated November 6, 1997, from M. Wayne Hodges, RES, to John T. Larkins, Executive Director, ACRS. Subject: Draft EMI/RFI Regulatory Package Containing DG-1029, Rev. 1.7.

cc: J. Hoyle, SECY
J. Blaha, OEDO
J. Mitchell, OEDO
M. Knapp, RES
S. Collins, NRR
T. Martin, AEOD

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

December 9, 1997

MEMORANDUM TO: L. Joseph Callan
Executive Director for Operations

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

SUBJECT: DRAFT REGULATORY GUIDE DG-1071, "STANDARD FORMAT
AND CONTENT FOR POST-SHUTDOWN DECOMMISSIONING
ACTIVITIES REPORT"

During the 447th meeting of the Advisory Committee on Reactor Safeguards, December 3-6, 1997, the Committee decided not to review the subject draft regulatory guide.

Reference:

U. S. Nuclear Regulatory Commission Draft Regulatory Guide DG-1071, "Standard Format and Content for Post-Shutdown Decommissioning Activities Report," September 1997.

cc: J. Hoyle, SECY
J. Blaha, OEDO
J. Mitchell, OEDO
S. Collins, NRR
J. Roe, NRR
S. Weiss, NRR
M. Masnik, NRR

**UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D C 20555
December 11, 1997**

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

Dear Chairman Jackson

SUBJECT PROPOSED FINAL REGULATORY GUIDE 1 174 AND STANDARD REVIEW PLAN
CHAPTER 19 FOR RISK-INFORMED, PERFORMANCE-BASED REGULATION

During the 446th and 447th meetings of the Advisory Committee on Reactor Safeguards, November 6-7 and December 3-6, 1997, respectively, we met with representatives of the NRC staff to review proposed final Regulatory Guide 1 174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Current Licensing Basis," and Standard Review Plan (SRP) Chapter 19 (General Guidance) for risk-informed, performance-based regulation. We discussed the staff's reconciliation of public comments on the subject documents, including proposed changes to address policy issues under consideration by the Commission. Our Subcommittee on Reliability and Probabilistic Risk Assessment (PRA) met with the staff and industry representatives on October 21-22 and November 12-13, 1997, to discuss these matters. We also had the benefit of the documents referenced.

Conclusions and Recommendations

- 1 We recommend that Regulatory Guide 1 174 and associated Standard Review Plan Chapter 19 be approved and issued for use by the industry and staff.
- 2 The modification of the acceptance guidelines to allow consideration of very small increases in CDF (core damage frequency) and LERF (large, early release frequency) for a broader range of the total CDF or LERF values is appropriate.
- 3 The decisionmaking process described in Regulatory Guide 1 174 and, in particular, its treatment of quantified and unquantified uncertainties, is sound. The staff has correctly focused on identifying the important sources of uncertainty and determining their impact on decisions, rather

than simply using the final distribution as the sole basis for decisionmaking

- 4 The staff discussion of PRA quality in these documents is appropriate. We agree with the staff position that the assessment of the scope and quality of the probabilistic analyses should focus on whether they are adequate for the purpose intended.

As we stated in our report dated March 17, 1997, we believe that this new process and these documents are a significant achievement that will contribute to the safe and efficient use of nuclear power.

We believe that these documents will evolve as experience is gained. We again urge the staff to seek innovative applications of the risk-informed approach to regulation so that this Regulatory Guide and the associated Standard Review Plan Chapter will be tested and improved upon in practice. We request the staff to brief the Committee periodically on this regulatory activity.

Sincerely,

R. L. Seale
Chairman

References.

- 1 Memorandum dated November 24, 1997, from M. Wayne Hodges, Office of Nuclear Regulatory Research, NRC, and Gary M. Holahan, Office of Nuclear Reactor Regulation, NRC, to John Larkins, ACRS, Subject: "General Regulatory Guide (DG-1061) and Standard Review Plan (SRP-Chapter 19) for Risk Informed Regulatory Decisionmaking for Plant Specific CLB Changes," with attachments, as follows:
 - Proposed Final Regulatory Guide 1.174 (Draft Guide DG-1061) dated November 25, 1997, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Current Licensing Basis."
 - Proposed Final Standard Review Plan Chapter 19, Revision N, dated November 25, 1997, "Use of Probabilistic Risk Assessment in Plant-Specific, Risk-Informed Decisionmaking: General Guidance."

- 3 -

- 2 Draft SECY dated November 7, 1997, from L Joseph Callan, Executive Director for Operations, NRC, for the Commissioners, "Final Regulatory Guidance on Risk-Informed Regulation Policy Issues "

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

March 17, 1997

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

Dear Chairman Jackson:

SUBJECT: PROPOSED STANDARD REVIEW PLAN SECTIONS AND REGULATORY
GUIDES FOR RISK-INFORMED, PERFORMANCE-BASED REGULATION

During the 437th, 438th, and 439th meetings of the Advisory Committee on Reactor Safeguards, December 5-7, 1996, February 6-8, and March 6-8, 1997, respectively, we met with representatives of the NRC staff, industry, and other interested parties to review the proposed Standard Review Plan (SRP) sections, Regulatory Guides, and other matters associated with risk-informed, performance-based regulation. We discussed the staff's approach to codify risk-informed, performance-based regulation into a general guidance SRP section and an associated Regulatory Guide, as well as related documents for technical specifications, inservice testing, and graded quality assurance. We also discussed industry views and initiatives related to these matters. In addition, our Subcommittee on Probabilistic Risk Assessment (PRA) met with the staff and industry representatives to discuss these documents on October 31, November 1, 21, 22, 1996, and January 28, February 20 and 21, 1997. We also had the benefit of the documents referenced.

Conclusions and Recommendations

1. The draft Regulatory Guide DG-1061 (General Guidance) and the associated SRP Chapter 19 that provide guidance for making risk-informed changes to the current licensing basis of individual plants constitute a significant achievement. They, and in particular the stated principles, provide the foundation for risk-informed regulatory philosophy that can better focus resources and can lead to a more coherent regulatory structure. These documents should be issued for public comment. The staff has formulated questions that should elicit the public comments needed to refine and improve these draft documents.
2. The draft Regulatory Guides for application of risk-informed decisionmaking to technical specifications (DG-1065) and inservice testing (DG-1062) have been developed consistent with

the principles articulated in the General Guidance (DG-1061). These Regulatory Guides and associated SRP sections should be issued for public comment. The proposed guidelines for acceptability of allowed outage time changes are based on the incremental conditional probability of core damage and large, early release. Such guidelines could also include limits on the maximum conditional annual core damage frequency (CDF) and large, early release frequency (LERF). This alternative is one of the questions included in the proposed Federal Register notice, and we believe that final resolution of this issue can be postponed until public comments on this matter have been received.

3. The version of the Regulatory Guide that we reviewed for application of risk-informed decisionmaking to graded quality assurance (DG-1064) took an unnecessarily timid approach toward focusing stringent quality assurance activities on highly risk-significant systems, structures, and components (SSCs) and defining adequate, but less exacting, quality assurance demands on areas of low risk significance. There should be a clear justification based on PRA or other compelling reasons for classifying SSCs as belonging to the high-safety-significant category. Issuance of this draft Regulatory Guide in this form for public comment could erroneously reinforce the wide-spread suspicion that risk-informed regulation is simply an additional layer of regulation imposing burden without tangible benefit. The staff is currently working to revise this document, and we are confident that the revised version will, in large measure, address our concerns. Since there is intense industry interest in this Regulatory Guide, we have no objection to the staff's proposal for issuing this document for public comment.
4. The successful implementation of the new regulatory philosophy will require a change in culture for both the NRC staff and the industry. A vigorous program should be established to communicate the risk-informed philosophy through workshops and other means planned by the staff.
5. The Commission should consider issuing a statement inviting licensees to propose new and innovative approaches to risk-informed, performance-based regulation using the concepts articulated in the General Guidance (DG-1061). The review and approval processes may need to be revised to provide timely responses to licensee submittals, which will necessarily cross disciplinary and organizational lines.
6. Appendix B of DG-1061 provides a method for estimating LERF in the absence of a Level 2 PRA. We recommend that some approaches also be developed for estimating the contributions

of external events to CDF and LERF, as well as from low-power and shutdown operations when detailed PRAs are not available.

Discussion

It has been about 22 years since the Reactor Safety Study (WASH-1400) introduced PRA to the reactor safety community. During this time, PRA methods, especially those for internal events during power operations, have matured to the point that PRA insights are increasingly being utilized in risk management both by the staff and licensees. However, formal guidance as to how PRA results can be used in the regulatory arena has been lacking. The documents that the staff has prepared provide such guidance. They constitute a major step forward in the development of a more risk-informed regulatory process.

Formulation of the guidance in terms of a basic set of principles creates a foundation for the new regulatory philosophy. We believe this to be a sound and significant achievement. It provides the starting point for the integration of traditional engineering approaches to safety, such as defense-in-depth, and the new probabilistic approach. The implementation of the General Guidance will evolve as experience is gained. We are confident, however, that a good start has been made.

The efforts to understand how the concepts of defense-in-depth and safety margins can be considered in the context of PRA must be applauded and encouraged. They provide very useful insights regarding the intent of these cornerstones of traditional reactor safety philosophy and the extent to which they are reflected in the PRA results.

We agree with the use of an "integrated" process in risk-management situations. It is clearly recognized that decisionmaking cannot rely solely on numerical results from either the PRA or more traditional approaches. We note that this integrated approach to decisionmaking is akin to the concept of the inclusion of "deliberation" in reaching risk-management decisions, as discussed in a recent report by the National Research Council Committee on Risk Characterization.

A first reading of the proposed SRP sections and associated Regulatory Guides creates the impression that they impose an onerous burden and are difficult to understand. We believe that the potential benefits clearly outweigh this burden and merit the effort to implement the new philosophy.

At this time, many of the staff and the industry may still not believe that risk-informed regulation is real or may have difficulty in making the transition to risk-informed decision-making. We are, therefore, very pleased to hear that the staff

plans to organize workshops and public meetings to explain the new regulatory philosophy.

Interaction with industry is needed to ensure that the industry realizes its responsibility to provide the staff with the information needed to make meaningful risk-informed decisions. To a large extent, the benefits of a risk-informed approach will be in proportion to the attention to accuracy and completeness of the industry's PRAs. There probably are licensees that have not yet done enough with their current PRAs to be able to garner significant benefit from a risk-informed approach to regulation. The new approach may, in the beginning, require additional industry effort. This is understandable and should be considered in the context of potential increases in safety and reductions in regulatory burden.

Graded quality assurance is a quintessential subject for application of risk-informed decisionmaking. Risk information should be the rational basis for adjudicating the level of quality assurance effort needed to provide confidence that SSCs will perform their safety functions reliably. The staff is currently planning to use risk information only to reduce quality assurance requirements for SSCs in the low-safety-significant category. We believe there is a better approach to using risk information to classify SSCs according to quality assurance needs than that described in the draft Regulatory Guide. Greater discrimination among the quality assurance needs will better focus licensee and regulatory attention on risk-important topics. Such a focus may not be achieved by simply using risk information to define reductions in licensee burdens associated with quality assurance for low-safety-significant items.

The approach proposed in Appendix B of DG-1061 for estimating LERF in the absence of a Level 2 PRA needs to be supported with additional documentation. Although this approach may be appropriate for screening purposes, additional probabilistic analyses using plant-specific values may be necessary for plants that do not meet the LERF guidelines.

We express our appreciation for the staff's cooperation during this long process. We have had excellent discussions of both concepts and methods during our meetings. The staff was always willing to listen and debate with us. The frequent interactions between the staff and the Committee were very valuable and constructive.

Sincerely,

A handwritten signature in dark ink, appearing to read "R. L. Seale". The signature is fluid and cursive, with the first letters of the first and last names being capitalized and prominent.

R. L. Seale
Chairman

References:

1. U.S. Nuclear Regulatory Commission, Draft Regulatory Guide DG-1061, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Current Licensing Basis," dated February 28, 1997.
2. U.S. Nuclear Regulatory Commission, Draft Regulatory Guide DG-1062, "An Approach for Plant Specific, Risk-Informed, Decision Making: Inservice Testing," dated February 25, 1997.
3. U.S. Nuclear Regulatory Commission, Draft Regulatory Guide DG-1064, Revision 4, "An Approach for Plant-Specific, Risk-Informed Decision Making: Graded Quality Assurance," dated February 26, 1997.
4. U.S. Nuclear Regulatory Commission, Draft Regulatory Guide DG-1065, Revision 5, "An Approach for Plant-Specific, Risk-Informed Decision Making: Technical Specifications," dated February 24, 1997.
5. U.S. Nuclear Regulatory Commission, Draft Standard Review Plan, Chapter 19, Revision L, "Use of Probabilistic Risk Assessment in Plant-Specific, Risk-Informed Decisionmaking: General Guidance," dated March 3, 1997.
6. U.S. Nuclear Regulatory Commission, Draft Standard Review Plan, Chapter 3.9.7, Revision 2C, "Standard Review Plan for the Review of Risk-Informed Inservice Testing Applications," dated February 25, 1997.
7. U.S. Nuclear Regulatory Commission, Draft Standard Review Plan, Chapter 16.1, Revision 12, "Risk-Informed Decision Making: Technical Specifications," dated February 24, 1997.
8. U. S. Nuclear Regulatory Commission, NUREG-75/014, "Reactor Safety Study, An Assessment of Accident Risks in U.S. Nuclear Power Plants," WASH-1400, October 1975.
9. National Research Council report, "Understanding Risk, Informing Decisions in a Democratic Society," 1996.

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

May 9, 1997

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

Dear Chairman Jackson:

SUBJECT: THE POLICY REGARDING STOCKPILING OF POTASSIUM IODIDE

During the 441st meeting of the Advisory Committee on Reactor Safeguards, May 1-3, 1997, we completed our discussion of the bases for the NRC staff's denial of a petition for rulemaking relating to the reevaluation of the policy regarding the use of potassium iodide (KI) after a severe accident at a nuclear power plant. During our 440th meeting, April 3-4, 1997, we discussed this matter with representatives of the NRC staff, the Nuclear Energy Institute (NEI), and the State of Illinois. We also had the benefit of the documents referenced.

Mr. Peter Crane, a member of the NRC staff in the Office of the General Counsel, had previously filed a differing professional opinion in 1989 requesting a reevaluation of the Commission's policy on KI. Mr. Crane also filed a petition on September 9, 1995, as a private citizen requesting a rulemaking to implement the recommendations of the Kemeny Commission that the United States stockpile KI for protection of the thyroid following nuclear accidents. Specifically, Mr. Crane requested that the Commission amend its regulations (10 CFR 50.47(b)(10)) to specify that the prophylactic use of KI for the general population within the plume exposure pathway Emergency Planning Zone (EPZ) for each licensed nuclear power plant be identified as one of the "range of protective actions" required to be set forth in State and local emergency plans.

There is no argument that KI can act as an effective radioiodine blocker if administered in a timely manner. But in the supporting documentation and presentations, the staff did not adequately address several technical concerns that have been expressed repeatedly when this issue has arisen in the past. These include the spectrum of side effects that would be encountered in the administration of KI to a large population or the identification of the timing of KI administration to ensure effective blocking of

radioiodine released during nuclear accidents. We saw no meaningful study of KI predistribution, nor did we see any examination of limited KI shelf life.

Conclusion

We find the arguments for the stockpiling of KI to be unconvincing. Therefore, we agree with the staff's position that a revision to the regulations is unwarranted and that the petition should be denied.

Discussion

In 1996, the Federal Radiological Preparedness Coordinating Committee (FRPCC) had convened an ad hoc Subcommittee on Potassium Iodide to review the public use of KI following a nuclear accident. Based on its evaluation of new information and comments from interested parties, the FRPCC Subcommittee concluded that "while the viewpoints presented at the public meeting were compelling, the 1996 Subcommittee heard no new information that seriously challenges the bases for the 1985 recommendation concerning public use of KI." The 1985 policy did not recommend predistribution or stockpiling of KI for the general public. Nonetheless, the FRPCC Subcommittee made several recommendations including the following:

- Without changing the Federal policy, and without interceding in the State's prerogative to make its own decision on whether or not to use KI, the Federal Government (NRC, or through FEMA, etc.) should fund the purchase of a KI stockpile for any State that, hereafter, decides to incorporate its use as a protective measure for the general public.
- Local jurisdictions, who have the option of incorporating the use of KI in their protective measures independent of the State's plan, consider, in consultation with the State, the use of KI as a possible protective measure, and be aware that if they choose to do so, they would then incur a responsibility to develop plans for distribution or predistribution of the KI.

The full FRPCC endorsed the Subcommittee's recommendations and plans to publish a revised Federal policy statement on the distribution of KI. The staff's position is that a revision to the regulations as requested by the petitioner is unnecessary because the anticipated actions by the FRPCC will address substantially the fundamental concerns of the petitioner without incurring the burden of changing all State and local emergency plans. The actions expected to be recommended by the FRPCC will ensure that, in contrast to the inadequate supply situation which existed at the time of the Three Mile Island accident, KI could be made available if needed.

Representatives of NEI and the State of Illinois concurred with the staff's recommendation that the petition be denied and that the State and local governments be allowed to decide when to include KI for general public use in emergency plans. They argued that stockpiling or predistributing KI for the general public will not add any significant public health and safety benefit beyond that provided by existing emergency preparedness at commercial nuclear power plants.

Additional comments by ACRS Member Dr. T. S. Kress are presented below.

Sincerely,

R. L. Seale
Chairman

Additional Comments by ACRS Member Dr. T. S. Kress

I disagree with the Committee's position on this issue. It has been recognized for more than 40 years that KI would be an effective prophylaxis against fission-product-iodine-induced thyroid cancer if taken on a timely basis before exposure. In the face of this almost universally agreed upon knowledge, the long-standing policy of NRC of leaving the decision up to the States as to whether or not to stockpile KI has resulted in only two States adopting the strategy. It is obvious that the desired result of having this remedy available if needed has not been achieved by this policy and most likely never will be.

The major difficulties with KI stockpiling are "limited shelf life" and "difficulty of distribution" (or whether or not to pre-distribute). The issue of limited shelf life essentially has been resolved by new encapsulation technology. The distribution issue just needs a real resolve on the part of NRC to address it. There are a number of workable solutions. The oft-cited failed initial attempt in Tennessee to predistribute should not be taken as an absolute indicator of the potential for that strategy. There have been no follow-up investigations as to why this attempt failed and how to make it work.

The federally controlled stockpiling of KI cannot and should not be justified on risk reduction or cost/benefit considerations. It should be viewed as being strictly defense-in-depth (as are other emergency response measures) and is justified purely on prudence grounds. NRC should actively work with the States to set up an effective system stockpiling and distributing KI.

References:

1. Memorandum dated March 26, 1997, from Peter Crane for the ACRS, transmitting Rulemaking Petition dated September 1995 on Potassium Iodide.
2. Memorandum dated February 13, 1997, from Bill M. Morris, Office of Nuclear Regulatory Research, NRC, to John T. Larkins, ACRS, Subject: ACRS Review of the Denial of Petition for Rulemaking (PRM-50-63) Relating to a Re-Evaluation of the Policy Regarding Use of Potassium Iodide after a Severe Accident at a Nuclear Power Plant.

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

September 10, 1997

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

Dear Chairman Jackson:

SUBJECT: STAFF ACTION PLAN TO IMPROVE THE SENIOR MANAGEMENT MEETING PROCESS

During the 444th meeting of the Advisory Committee on Reactor Safeguards, September 3-5, 1997, we met with representatives of the NRC staff to discuss its Action Plan to improve the Senior Management Meeting (SMM) process. Our Subcommittees on Probabilistic Risk Assessment, Plant Operations, and Fire Protection also discussed this matter during a joint meeting on August 28-29, 1997. We also had the benefit of the documents referenced.

The SMM process is being revised in response to Commission direction. A report prepared by Arthur Andersen contained recommendations for improving the SMM process in two areas: the SMM information base and the SMM evaluation process. The first area for improvement involves inputs to the SMM decisionmaking process, including performance indicators and the decision criteria used by the senior managers. The second area involves the role of SMM participants, the method of reaching consensus, the presentation of information, and the documentation of meeting results.

The objectives of the revised SMM process are: to provide more structure to the performance evaluations, increase participation of senior managers, improve consistency among the Regions, and enhance the scrutability of the process and decisions to both the Commission and the public. In addition, the Commission directed the staff to make further improvements to the SMM process by developing better performance indicators that can provide a more objective basis for judging whether a nuclear power plant licensee can be placed on or be removed from the NRC Watch List. These improved indicators and objective measures are expected to enhance the staff's ability to take appropriate regulatory actions, including additional enforcement measures - some of which have not been effective in the past.

The staff presented to us its plan to improve the SMM information flow, in order to obtain objective data for use in the assessment process. A key element in this improved process is the development of a Performance Template which is designed to coordinate all relevant data to improve decisionmaking.

We support the goal of codifying the SMM information gathering and review process, however, the basis for the top-level criteria contained in the Template is not clear. Furthermore, the process by which the Template leads to the formulation of decisions is also not apparent.

We would prefer to see a top-down structure that starts with the point of decision, identifies the objectives of the decision, and proceeds to define the informational needs. For example, in a risk-informed approach, one could envision as an objective (one of several) the prevention of the occurrence of initiating events and the degradation of safety functions. To satisfy this objective, one would look for precursors to these undesirable events and then would proceed to identify relevant performance indicators, and address the issue of how these would be measured. In this way, the logic behind the Template would be transparent and easy to communicate to the various stakeholders. A similar systematic approach would be taken for the other objectives. The staff told us that such a hierarchical structure will be developed. We recommend that its development be accelerated, and we would like to be kept informed.

In addition to the concern with the Template discussed above, we recommend that the staff address the following items regarding its efforts to improve the SMM information base:

- Examination of a sample of significant operational events is needed to determine if they were foreshadowed by prior plant performance.
- Careful assessment of the pros and cons on the use of economic indicators is needed, as the relationship between economic indicators and safety performance is not clearly understood.
- Evaluation of how the revised process will focus on the competency of plant management and culture is needed.
- Assurance is needed that the new performance standards are objective and reduce reliance on event-driven assessments.

We note that although progress has been made in improving the information base of the SMM, considerable work remains in such areas as the development of tools for assessing management/organizational effectiveness and testing their implementation.

The staff also discussed the status of its plans to perform an integrated review of the NRC assessment process. The Office of Nuclear Reactor Regulation (NRR) has a number of programs in place to assess licensee performance. Among these are the Systematic Assessment of Licensee Performance, the SMM process, the Plant Performance Review, and the Plant Issues Matrix. Each of these programs provides insight on some aspects of licensee performance. Currently, there is no integrated assessment of licensee performance. The NRR staff is undertaking a review and examination of its current programs with the intention of identifying improvements that will provide a better, integrated, and more comprehensive assessment of licensee performance. Development of a hierarchical structure similar to the one recommended above would be useful here.

The staff plans to complete its integrated review of the assessment process by March 1998, and provide recommendations to the Commission by June 1998. The staff has not yet defined the requirements (preferably quantitative) for an adequate program to assess licensee performance. It is not apparent to us how well-designed recommendations can be formulated without the explicit definition of the requirements for the assessment program to meet Agency needs. It is not clear how preferred options can be selected absent explicit requirements. We strongly recommend that NRR develop requirements for an adequate licensee performance assessment program.

We plan to meet with the NRC staff as it continues its integrated review of the NRC assessment process for operating commercial nuclear power plants.

Sincerely,

R. L. Seale
Chairman

References:

1. SECY-97-072, Memorandum dated April 2, 1997, from L. Joseph Callan, Executive Director for Operations, NRC, for the Commissioners, Subject: Staff Action Plan to Improve the Senior Management Meeting Process.
2. SECY-97-192, Memorandum dated August 21, 1997, from L. Joseph Callan, Executive Director for Operations, NRC, for the Commissioners, Subject: Peer Review of the Arthur Andersen Methodology and Use of Trending Letters.
3. U.S. Nuclear Regulatory Commission, Senior Management Meeting (SMM), Directive 8.14, Volume 8: Licensee Oversight Programs, Approved March 19, 1997.
4. Memorandum dated March 14, 1997, from John C. Hoyle, Secretary, NRC, to L. Joseph Callan, Executive Director for

Operations, NRC, Subject: Staff Requirements - Briefing on Analysis of Quantifying Plant Watch List Indicators (Arthur Andersen Study), Commissioners Conference Room, February 18, 1997.

5. Memorandum dated June 24, 1997, from John C. Hoyle, Secretary, NRC, to L. Joseph Callan, Executive Director for Operations, NRC, and John T. Larkins, ACRS/ACNW, Subject: Staff Requirements - Briefing on Staff Response to Arthur Andersen Study Recommendations, April 24, 1997, Commissioners Conference Room.
6. Memorandum dated June 30, 1997, from John C. Hoyle, Secretary, NRC, to L. Joseph Callan, Executive Director for Operations, NRC, Subject: Staff Requirements - Briefing on Operating Reactors and Fuel Facilities, June 25, 1997, Commissioners Conference Room.
7. Memorandum dated August 19, 1997 from John C. Hoyle, Secretary, NRC, to L. Joseph Callan, Executive Director for Operations, NRC, and John T. Larkins, ACRS/ACNW, Subject: Staff Requirements - SECY-97-122 - Integrated Review of the NRC Assessment Process for Operating Commercial Nuclear Reactors.

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

February 12, 1997

MEMORANDUM TO: Hugh L. Thompson, Jr.
Acting Executive Director for Operations
FROM: John T. Larkins, Executive Director
Advisory Committee on Reactor Safeguards
SUBJECT: RULEMAKING PLAN TO AMEND FINANCIAL
PROTECTION REQUIREMENTS FOR PERMANENTLY
SHUTDOWN NUCLEAR POWER REACTORS --
10 CFR 50.54(W) AND 10 CFR 140.11

During the 438th meeting of the Advisory Committee on Reactor Safeguards, February 6-8, 1997, the Committee decided not to review the subject rulemaking.

Reference:

SECY-96-256, Memorandum dated December 17, 1996, from James M. Taylor, Executive Director for Operations, for the Commissioners, Subject: Changes to the Financial Protection Requirements for Permanently Shutdown Nuclear Power Reactors, 10 CFR 50.54(W) and 10 CFR 140.11

cc: J. Hoyle, SECY
J. Blaha, OEDO
J. Mitchell, OEDO
D. Morrison, RES
F. Miraglia, NRR
G. Mencinsky, RES
J. Cortez, RES

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

February 20, 1997

MEMORANDUM TO: Leonard J. Callan
Executive Director for Operations

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

SUBJECT: PROPOSED RULEMAKING FOR AMENDMENTS TO 10 CFR
PARTS 30, 40, 50, 70, AND 72 -- SELF-GUARANTEE
FOR NON-PROFIT AND NON-BOND ISSUING LICENSEES

During the 438th meeting of the Advisory Committee on
Reactor Safeguards, February 6-8, 1997, the Committee decided not
to review the subject rulemaking.

Reference:

Memorandum dated December 19, 1996, from David L. Morrison,
Office of Nuclear Regulatory Research, to NRC Office Directors,
Subject: Office Review and Concurrence on a Proposed Rulemaking
for Amendments to 10 CFR Parts 30, 40, 50, 70, and 72 -- Self-
Guarantee for Non-Profit and Non-Bond Issuing Licensees

cc: J. Hoyle, SECY
J. Blaha, OEDO
J. Mitchell, OEDO
D. Morrison, RES
C. Prichard, RES
J. Cortez, RES

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

March 25, 1997

MEMORANDUM TO: L. Joseph Callan
Executive Director ~~for Operations~~

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

SUBJECT: PROPOSED AMENDMENTS TO 10 CFR PART 55, "INITIAL
LICENSED OPERATOR EXAMINATION REQUIREMENTS"

During the 439th meeting of the Advisory Committee on Reactor Safeguards, March 6-8, 1997, the Committee decided to review the proposed amendments to 10 CFR Part 55 after reconciliation of public comments. The Committee has no objection to the issuance of these proposed amendments for public comment.

Reference:

Memorandum dated March 4, 1997, from David L. Morrison, Office of Nuclear Regulatory Research, to NRC Office Directors, Subject: Proposed Rule - Initial Licensed Operator Examination Requirements, Amendments to 10 CFR Part 55.

cc: J. Hoyle, SECY
J. Blaha, OEDO
J. Mitchell, OEDO
D. Morrison, RES
H. Tovmassian, RES
J. Cortez, RES
S. Guenther, NRR

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

April 8, 1997

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

Dear Chairman Jackson:

SUBJECT: PROPOSED REGULATORY GUIDANCE RELATED TO IMPLEMENTATION OF
10 CFR 50.59 (CHANGES, TESTS AND EXPERIMENTS)

During the 440th meeting of the Advisory Committee on Reactor Safeguards, April 3-4, 1997, we met with representatives of the NRC staff and the Nuclear Energy Institute (NEI) regarding SECY-97-035, "Proposed Guidance Related to Implementation of 10 CFR 50.59 (Changes, Tests and Experiments)." We discussed the staff's approach to clarifying guidance for implementing 10 CFR 50.59 and proposed options for resolving policy issues.

Conclusions and Recommendations

We recommend that SECY-97-035, as now formulated, not be issued for public comment. We recommend, instead, additional NRC and industry interaction regarding this matter before the proposed guidance in SECY-97-035 is issued for public comment.

Discussion

The industry and staff have over 30 years of experience in implementing 10 CFR 50.59. Over this time, the staff has identified concerns in only a small subset of situations evaluated under 10 CFR 50.59. In SECY-97-035, the staff stated the following with regard to the current process and industry implementation of NSAC-125:

Although the staff has not endorsed NSAC-125, it has concluded, as discussed in the April 15, 1996, memorandum from James M. Taylor to Chairman Jackson, that NSAC-125 has given the nuclear power industry a reasonable foundation to establish a process that will, in most instances, produce effective evaluations related to changes to plant design or procedures. Changes of significance are highly likely to be identified by the licensee through implementation of the NSAC-125 guidance.

Inspection results have confirmed that the quality of the evaluations of changes has improved since licensees began implementing the NSAC-125 guidance. However, the NSAC-125 guidance is not a requirement for any licensee, and each licensee develops its own program for performing the required evaluations under 10 CFR 50.59.

The staff also found that difficulties arise in the licensee's day-to-day use of the 10 CFR 50.59 process when the staff and licensee have a different understanding and different expectations for implementation of the rule. The staff, therefore, is proposing additional regulatory guidance in SECY-97-035 to reduce the potential for deficiencies in implementing 10 CFR 50.59. Since the staff appears to agree that when the NSAC-125 guidance has been implemented properly it has generally resulted in satisfactory safety evaluations, it would seem more effective to work with the industry to build on NSAC-125. The goal would be for the staff to endorse an appropriate version of NSAC-125 with exceptions, as needed. It is our understanding that the industry has attempted to improve on NSAC-125 through the development of draft guideline NEI 96-07, "Guidelines for 10 CFR 50.59 Safety Evaluations." These improvements may well address many of the present concerns.

Sincerely,

R. L. Seale
Chairman

References:

1. SECY-97-035, Memorandum dated February 12, 1997, from H. L. Thompson, Jr., Acting Executive Director for Operations, NRC, for the Commissioners, Subject: Proposed Regulatory Guidance Related to Implementation of 10 CFR 50.59 (Changes, Tests and Experiments).
2. Memorandum dated April 15, 1996, from James M. Taylor, Executive Director for Operations, NRC, to Shirley Ann Jackson, Chairman, NRC, Subject: Action Plan for Improvements to 10 CFR 50.59 Implementation and Oversight.
3. Electric Power Research Institute, Nuclear Safety Analysis Center, NSAC-125, "Guidelines for 10 CFR 50.59 Safety Evaluations," June 1989.
4. Nuclear Energy Institute, NEI 96-07, draft Revision A, "Guidelines for 10 CFR 50.59 Safety Evaluations," July 1996.

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

April 8, 1997

MEMORANDUM TO: L. Joseph Callan
Executive Director *for Operations*
John T. Larkins
FROM: John T. Larkins, Executive Director
ACRS/ACNW
SUBJECT: PROPOSED RULEMAKING PLAN FOR AMENDMENTS TO
SUBPART H AND APPENDIX A TO 10 CFR PART 20 --
RESPIRATORY PROTECTION

During the 440th meeting of the Advisory Committee on
Reactor Safeguards, April 3-4, 1997, the Committee decided not to
review the subject rulemaking plan.

Reference:

Memorandum dated March 24, 1997, from David L. Morrison, Office
of Nuclear Regulatory Research, NRC, to Addressees, Subject:
Rulemaking Plan: Respiratory Protection (Subpart H and Appendix
A to Part 20)

cc: J. Hoyle, SECY
J. Blaha, OEDO
J. Mitchell, OEDO
D. Morrison, RES
A. Roecklein, RES
J. Cortez, RES

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

April 8, 1997

MEMORANDUM TO: L. Joseph Callan
Executive Director for Operations

FROM: John T. Larkins, *John T. Larkins*
Executive Director
ACRS/ACNW

SUBJECT: PROPOSED RULE - FINANCIAL ASSURANCE
REQUIREMENTS FOR DECOMMISSIONING
NUCLEAR POWER REACTORS

During the 440th meeting of the Advisory Committee on Reactor Safeguards, April 3-4, 1997, the Committee decided not to review the subject proposed rule.

Reference:

Memorandum dated March 10, 1997, from David L. Morrison, Office of Nuclear Regulatory Research, NRC, to Addressees, Subject: Office Review and Concurrence on a Proposed Rule - Financial Assurance Requirements for Decommissioning Nuclear Power Reactors

cc: J. Hoyle, SECY
J. Blaha, OEDO
J. Mitchell, OEDO
D. Morrison, RES
B. Richter, RES
J. Cortez, RES

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

April 11, 1997

MEMORANDUM TO: L. Joseph Callan
Executive Director for Operations

FROM: John T. Larkins, *John T. Larkins*
Executive Director
ACRS/ACNW

SUBJECT: PROPOSED RULEMAKING PLAN FOR EMERGENCY
PLANNING REQUIREMENTS FOR PERMANENTLY
SHUTDOWN NUCLEAR POWER PLANT SITES

During the 440th meeting of the Advisory Committee on Reactor Safeguards, April 3-4, 1997, the Committee decided not to review the subject rulemaking plan.

Reference:

Memorandum dated March 11, 1997, from David L. Morrison, Office of Nuclear Regulatory Research, NRC, to Addressees, Subject: Request for Concurrence on Rulemaking Plan for Emergency Planning Requirements for Permanently Shutdown Nuclear Power Plant Sites

cc: J. Hoyle, SECY
J. Blaha, OEDO
J. Mitchell, OEDO
D. Morrison, RES
G. Mencinsky, RES
J. Cortez, RES

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

May 8, 1997

Mr. L. Joseph Callan
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Mr. Callan:

SUBJECT: PROPOSED FINAL POLICY STATEMENT ON THE RESTRUCTURING
AND ECONOMIC DEREGULATION OF THE ELECTRIC UTILITY
INDUSTRY

During the 441st meeting of the Advisory Committee on Reactor Safeguards, May 1-3, 1997, we met with representatives of the NRC staff to discuss the draft final version of the NRC Policy Statement on the Restructuring and Economic Deregulation of the Electric Utility Industry. We also had the benefit of the documents referenced.

Conclusions and Recommendations

- The Policy Statement is an important part of the NRC Action Plan on Industry Deregulation and Utility Restructuring and will help to ensure adequate funding for operations and decommissioning of nuclear power plants.
- The issues associated with ownership of a U.S. nuclear power plant by a foreign entity should be reviewed so that the Commission will be better prepared to advise on the need for enabling legislation.
- The language in the Policy Statement pertaining to the assignment of financial liability for decommissioning costs should be clarified.
- The overall Action Plan should give consideration to the following potential safety concerns:
 - (1) reductions in manpower and training expenditures that could impact safe plant operations,
 - (2) cost-driven changes in plant operations that could significantly reduce safe operating margins,

- (3) inappropriate use of on-line maintenance that could result in a reduction of safety margins, and
- (4) undue erosion of fuel rod integrity margins as burnup increases.

Discussion

As the electric utility industry moves toward deregulation, there may be more premature shutdowns of nuclear power plants as individual utilities seek to minimize costs and reduce financial risk. Recent experience has shown that even some plants with good operating records and in good material condition will be permanently shut down if the owners believe that generating costs exceed the projected market price of electricity.

The concerns regarding adequate funding for decommissioning and an erosion of plant safety brought about by the market pressures of reducing operational costs need to be addressed. There is little experience with the decommissioning of nuclear power plants and the costs associated with disposal of spent fuel and low-level waste could have a significant impact on the total decommissioning costs. Cost estimates for decommissioning should be revised as experience is gained. It is encouraging that, as a few States have moved toward some form of retail electric power deregulation, most Public Utility Commissions have recognized that the issue of decommissioning is a matter of public health and safety and have allowed the recovery of decommissioning costs.

The policy of the NRC is to assign financial liability to licensees for decommissioning prorated according to fractional ownership. The NRC does reserve the right to impose "joint and several" financial liability on co-owners of nuclear power plants. In the Policy Statement, the NRC should make it clear that the agency will resort to "joint and several" liability only in an extreme situation where adequate protection of the public health and safety would otherwise not be maintained.

We would like to be kept informed on the resolution of our concerns associated with the NRC Action Plan pertaining to the restructuring and deregulation of the electric utility industry.

Sincerely,

R. L. Seale
Chairman

References:

1. Memorandum dated April 3, 1997 from T. T. Martin, Office of Nuclear Reactor Regulation, NRC, to J. T. Larkins, ACRS, transmitting Draft Final NRC Policy Statement on the Restructuring and Economic Deregulation of the Electric Utility Industry (Predecisional Draft).
2. Draft memorandum (undated) from L. J. Callan, Executive Director for Operations, NRC, to the Commissioners, Subject: Proposed Rule on Financial Assurance Requirements for Decommissioning Nuclear Power Reactors.
3. Office of Nuclear Reactor Regulation, Excerpt from Director's Bimonthly Status Report, March 1997, on Industry Deregulation and Utility Restructuring Action Plan (NRC Internal Document).
4. Public Comments Received on draft Policy Statement on the Restructuring and Economic Deregulation of the Electric Utility Industry, undated.

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

June 19, 1997

MEMORANDUM TO: L. Joseph Callan
Executive Director for Operations

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

SUBJECT: PROPOSED REGULATORY GUIDANCE RELATED TO IMPLEMENTATION OF 10 CFR 50.59 (CHANGES, TESTS AND EXPERIMENTS)

During the 442nd meeting of the Advisory Committee on Reactor Safeguards, June 11-14, 1997, the Committee discussed your May 5, 1997 response to the ACRS report dated April 8, 1997, to Chairman Jackson regarding proposed regulatory guidance related to implementation of 10 CFR 50.59 (Changes, Tests and Experiments). The Committee decided to continue its review of this matter after reconciliation of public comments.

Reference:

Letter dated May 5, 1997, from L. Joseph Callan, EDO, to R. L. Seale, Chairman, ACRS, Subject: Proposed Regulatory Guidance Related to Implementation of 10 CFR 50.59 ("Changes, Tests, and Experiments").

cc: J. Hoyle, SECY
J. Blaha, OEDO
J. Mitchell, OEDO
S. Collins, NRR
M. Slosson, NRR
A. Thadani, RES
J. Cortez, RES
D. Ross, AEOD

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

June 30, 1997

MEMORANDUM TO: L. Joseph Callan
Executive Director for Operations

FROM: John T. Larkins, Executive Director
ACRS/ACNW

SUBJECT: PROPOSED RULE, "FREQUENCY OF REVIEWS AND
AUDITS FOR EMERGENCY PREPAREDNESS PROGRAMS,
SAFEGUARDS CONTINGENCY PLANS, AND SECURITY
PROGRAMS FOR NUCLEAR POWER REACTORS," (10
CFR PARTS 50 AND 73, PRM 50-59 AND 50-60)

During the 442nd meeting of the Advisory Committee on
Reactor Safeguards, June 11-14, 1997, the Committee decided not
to review the subject proposed rule.

Reference:

Memorandum dated May 22, 1997, from David L. Morrison, Office of
Nuclear Regulatory Research, NRC, to Addressees, Subject: Office
Review and Concurrence on a Proposed Rule, "Frequency of Reviews
and Audits for Emergency Preparedness Programs, Safeguards
Contingency Plans, and Security Programs for Nuclear Power
Reactors," (10 CFR Parts 50 and 73, PRM 50-59 and 50-60).

cc: J. Hoyle, SECY
J. Blaha, OEDO
J. Mitchell, OEDO
A. Thadani, RES
S. Frattali, RES
S. Bahadur, RES
L. Bush, NRR
L. Cunningham, NRR

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

July 14, 1997

Mr. L. Joseph Callan
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Mr. Callan:

SUBJECT: PROPOSED FINAL REVISIONS TO 10 CFR PART 26, FITNESS-FOR-DUTY PROGRAM REQUIREMENTS

During the 443rd meeting of the Advisory Committee on Reactor Safeguards, July 9-11, 1997, we met with representatives of the NRC staff to discuss the proposed final modifications to the Fitness-For-Duty Program requirements, as well as the staff's resolution of the public comments. We also had the benefit of the documents referenced.

Overall, the proposed final modifications will strengthen the Fitness-For-Duty Program requirements and provide increased clarity and consistency without adding undue burden on the licensees. The proposed modifications will align the Fitness-For-Duty Program with the recommendations of the Department of Health and Human Services and reduce the possibility of subversion of test results. Therefore, we recommend issuance of the proposed final revisions to 10 CFR Part 26.

We encourage the staff to continue its work on exploring the benefits of Fitness-For-Duty performance indicators in creating a more effective and efficient approach for evaluating and regulating the Fitness-For-Duty Program.

Sincerely,

A handwritten signature in black ink, reading "R. L. Seale".

Robert L. Seale
Chairman

References:

Memorandum dated June 24, 1997, from A. Singh, ACRS, to John J. Barton, ACRS, transmitting the following:

1. U. S. Nuclear Regulatory Commission, Listing of Recommended Modifications to Proposed Revisions to 10 CFR Part 26, dated June 16, 1997.
2. Attachment A, Listing of Potential Modifications to Proposed Revisions to 10 CFR Part 26 that are not Recommended.
3. Attachment B, Text of Rule Showing Recommended Changes.
4. U. S. Nuclear Regulatory Commission, Draft NUREG, "Fitness for Duty in the Nuclear Industry: Responses to 1996 Public Comments," dated June 9, 1997.

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

July 15, 1997

MEMORANDUM TO: L. Joseph Callan
Executive Director for Operations

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

SUBJECT: PROPOSED DIRECT FINAL RULEMAKING: CHANGES TO
PARAGRAPH (h) OF 10 CFR 50.55a, "CODES AND
STANDARDS"

*R. Savio
for.*

During the 443rd meeting of the Advisory Committee on Reactor Safeguards, July 9-11, 1997, the Committee considered the subject rulemaking. The Committee plans to review the changes, if any, to this rule resulting from the resolution of public comments.

Reference:

Memorandum dated June 24, 1997, from Ashok Thadani, Director, RES, to John T. Larkins, Executive Director, ACRS, and Denwood Ross, Chairman, CRGR, Subject: Direct Final Rulemaking: Changes to Paragraph (h) of 10 CFR 50.55a, "Codes and Standards."

cc: J. Hoyle, SECY
J. Blaha, OEDO
J. Mitchell, OEDO
A. Thadani, RES
L. Shao, RES
M. Mayfield, RES
J. Cortez, RES
S. Collins, NRR
B. Boger, NRR
J. Wermiel, NRR
D. Ross, AEOD

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

September 10, 1997

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

Dear Chairman Jackson:

SUBJECT: PROPOSED RULEMAKING FOR SHUTDOWN AND FUEL STORAGE POOL
OPERATIONS AT NUCLEAR POWER PLANTS

During the 444th meeting of the Advisory Committee on Reactor Safeguards, September 3-5, 1997, we reviewed SECY-97-168, "Issuance for Public Comment of Proposed Rulemaking Package for Shutdown and Fuel Storage Pool Operation." During this review, we had the benefit of discussions with representatives of the NRC staff and the Nuclear Energy Institute. We also had the benefit of the documents referenced.

In our April 18, 1997 report to the Commission, we expressed concerns about the existing understanding of risk during shutdown operations. The NRC staff has recognized that shutdown operations may pose substantial risks. The staff sees a need for enforceable rules to ensure safety during shutdown operations. We note that, on numerous occasions in the past, the staff has found enforcement means when deficiencies in shutdown operations have been encountered. It is true, however, that there is no coherent set of requirements now available for the staff to inspect and monitor licensee activities during shutdown operations. This situation may not be relieved by a contrived interpretation of existing rules or unusual interpretations of conventional terms.

The nuclear industry has also recognized the importance of both safety and efficiency during shutdown operations. The industry has established cost-effective practices to achieve levels of safety during shutdown operations. These practices would be acceptable to the staff if it could enforce, inspect, and monitor these practices. Indeed, the proposed rule is a codification of a minimum subset of practices with aims similar to current industry practices.

The staff has been handicapped in the formulation of a rule for shutdown operations by an incomplete understanding of risk during this operating mode. Consequently, risk estimates made by the

staff in its regulatory analysis in support of the proposed rule are largely judgmental and lack the strong technical justifications that are available to estimate risks during power operations. The proposed rule would be vulnerable to a hostile or unsympathetic review.

The regulatory analysis guidelines constrain the staff from giving any credit for benefits of voluntary actions by the industry. As a result, the estimated risks of shutdown accidents for the purposes of a regulatory analysis should not be taken as an indicator of the real risks, which are likely to be considerably lower.

Many of the benefits attributed by the staff to elements of the proposed rule are of the intangible variety associated with enforceability and cannot be quantified in a meaningful way. The requirements in the proposed rule for fuel storage pool operations have benefits that are almost entirely of this intangible variety and provide no real safety benefit. We see little reason to burden an already difficult rulemaking effort with these requirements for fuel storage pool operations that provide so little safety benefit.

The staff has included requirements for fire protection in the proposed rule. There is, indeed, widespread agreement that fire protection is an essential safety issue during shutdown operations. It is not evident to us that existing fire protection rules are not applicable to shutdown operations and there is a need to augment these rules. We do recognize that the staff is committed to revise the fire protection regulations. It may not be opportune to add another element to the existing fire protection requirements only to modify these requirements in the near future.

We do believe shutdown operations of nuclear power plants deserve regulatory attention. Such regulatory attention is, in fact, overdue in light of the number of incidents of risk significance that have occurred in recent years and the heightened regulatory activity devoted to shutdown events. Coherent, risk-informed, enforceable requirements for shutdown operations that could be inspected and monitored in a consistent predictable manner would serve the interests of both the industry and the public. Seldom, however, has there been a clearer case for the cooperative development of such requirements by the industry and the staff. It is clear that it is possible to achieve a level of safety acceptable to the NRC staff at a cost palatable to the industry. Indeed, this situation may already exist absent only the elements of enforcement capability and bases for inspection and monitoring.

We find no merit in the suggestion that the Maintenance Rule (10 CFR 50.65) should be the basis for regulation of shutdown operations. We find compelling the view expressed by a representative of the Office of General Counsel that the

Maintenance Rule serves to assure that systems, components, and structures are capable of performing their safety functions. The rule cannot be construed to specify these safety functions.

We do not recommend issuance of the proposed rule on shutdown and fuel storage pool operations for public comment. We do recommend that the NRC staff explore the flexibilities of its policies and practices so it can work with industry to find ways to achieve enforceability and support for inspecting and monitoring the voluntary and apparently effective industry practices without impugning or penalizing such practices. We reiterate our belief that the staff needs to develop a more quantitative understanding of risk during all phases of low-power and shutdown operations. We further recommend that fuel storage pool operations not be included in the reexamination of the regulation of shutdown operations. Similarly, fire protection requirements for shutdown operations may be deferred to the more comprehensive reexamination of the existing fire protection regulations.

Sincerely,

R. L. Seale
Chairman

References:

1. SECY-97-168, Memorandum dated July 30, 1997, for the Commissioners, from L. Joseph Callan, Executive Director for Operations, NRC, Subject: Issuance for Public Comment of Proposed Rulemaking Package for Shutdown and Fuel Storage Pool Operation.
2. U. S. Nuclear Regulatory Commission, "Regulatory Analysis for the Proposed Regulation § 50.67, 'Shutdown and Fuel Storage Pool Operations at Nuclear Power Plants,'" July 24, 1997.
3. U. S. Nuclear Regulatory Commission, Draft Regulatory Guide DG-1066, "Shutdown and Fuel Storage Pool Operations at Nuclear Power Plants," July 24, 1997.
4. Report dated April 18, 1997 from R. L. Seale, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: Establishing a Benchmark on Risk During Low-Power and Shutdown Operations.

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

October 9, 1997

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

Dear Chairman Jackson:

SUBJECT: PROPOSED CHANGES TO 10 CFR 50.59 AND PROPOSED REVISION 1
TO GENERIC LETTER 91-18

During the 445th meeting of the Advisory Committee on Reactor Safeguards, October 2-3, 1997, we met with representatives of the NRC staff and the Nuclear Energy Institute (NEI) to discuss SECY-97-205, "Integration and Evaluation of Results From Recent Lessons-Learned Reviews," which includes proposed changes to 10 CFR 50.59 (Changes, Tests and Experiments) and Revision 1 to Generic Letter 91-18, "Information to Licensees Regarding NRC Inspection Manual Section on Resolution of Degraded and Nonconforming Conditions." We also discussed the proposed industry guidance document NEI 96-07, "Guidelines for 10 CFR 50.59 Safety Evaluations." We had the benefit of the documents referenced.

Conclusions and Recommendations

1. We recommend that the NRC issue Revision 1 to Generic Letter 91-18, since it explicitly clarifies the applicability of 10 CFR 50.59 evaluation process to address degraded and nonconforming conditions.
2. Because the current legal interpretation of 10 CFR 50.59 is at variance with past staff and industry practices, rulemaking appears to be necessary.
3. The staff should continue to work with NEI to reconcile NEI 96-07 with the staff's position rather than developing

separate guidance for implementing the 10 CFR 50.59 process. We recommend that the NRC endorse this industry approach with appropriate exceptions and clarifications.

4. We encourage the continued development of a plan for a 10 CFR 50.59 process that is consistent with risk-informed, performance-based regulation.

Discussion

In our April 8, 1997 report to the Commission, we recommended that the proposed guidance related to implementation of 10 CFR 50.59, as described in SECY-97-035, not be issued for public comment. Instead, we recommended that the NRC work with the industry to build on the guidance contained in NSAC-125. Our recommendation was based on consideration of over 30 years of industry experience, during which the staff identified problems in only a very small number of situations evaluated under 10 CFR 50.59.

Because the legal interpretation of 10 CFR 50.59 is at variance with past staff and industry practices, rulemaking appears to be necessary. However, rather than developing new regulatory guidance to support the current rule, the staff should issue a safety evaluation report or regulatory guide endorsing the guidance in the revised NEI 96-07 document. Any provisions in NEI 96-07 that the staff finds unacceptable could be identified as exceptions to NRC's acceptance of the industry guidance. This would be similar to past NRC practices of endorsing industrial standards subject to certain exceptions and clarifications.

The debate spawned by the proposed changes to 10 CFR 50.59 is indicative of the need to accelerate the move to risk-informed, performance-based regulation. The current 10 CFR 50.59 requirements already implement a form of this regulatory philosophy but at a very detailed level and in a manner that is inconsistent with current risk-management technology. Ideally, the performance requirements would be identified at a system or function level, and the licensees would have flexibility to manage the plants so long as these performance requirements are met (i.e., they stay within the defined envelope). Defining such performance requirements in advance would eliminate the present disagreements over whether "small" or "zero" risk increases are allowed.

The staff outlined a plan designed to enhance NRC oversight of licensee activities and to improve the existing regulatory process during the transition period to a more risk-informed, performance-based regulatory framework. In the interim, the industry needs to know whether it has a method acceptable to the NRC for performing proper safety evaluations per 10 CFR 50.59. We were informed by representatives of NEI that the industry is currently reviewing NEI 96-07, Revision 0, and that it is expected licensees will uniformly accept this guidance for performing safety evaluations.

Sincerely,

R. L. Seale
Chairman

References:

1. SECY-97-205, Memorandum dated September 10, 1997, from L. Joseph Callan, Executive Director for Operations, NRC, for the Commissioners, Subject: Integration and Evaluation of Results from Recent Lessons-Learned Reviews.
2. Draft NRC Generic Letter 91-18, Revision 1, "Information to Licensees Regarding NRC Inspection Manual Section on Resolution of Degraded and Nonconforming Conditions," September 1997.
3. SECY-97-035, Memorandum dated February 12, 1997, from Hugh L. Thompson, Jr., Acting Executive Director for Operations, NRC, for the Commissioners, Subject: Proposed Regulatory Guidance Related to Implementation of 10 CFR 50.59 (Changes, Tests and Experiments).
4. U.S. Nuclear Regulatory Commission, Draft NUREG-1606, "Proposed Regulatory Guidance Related to Implementation of 10 CFR 50.59 (Changes, Tests and Experiments)," April 1997.
5. Letter dated July 21, 1997, from Ralph E. Beedle, Nuclear Energy Institute, to Frank J. Miraglia, Jr., NRC, regarding NEI 96-07, Final Draft, Subject: Guidelines for 10 CFR 50.59 Safety Evaluations.
6. Report dated April 8, 1997, from R. L. Seale, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: Proposed

Regulatory Guidance Related to Implementation of 10 CFR 50.59
(Changes, Tests and Experiments).

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

October 15, 1997

MEMORANDUM TO: L. Joseph Callan
Executive Director for Operations

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

SUBJECT: PROPOSED FINAL AMENDMENTS TO 10 CFR PART 73,
"CHANGES TO NUCLEAR POWER PLANT SECURITY
REQUIREMENTS"

During the 445th meeting of the Advisory Committee on Reactor Safeguards, October 2-3, 1997, the Committee decided not to review the subject proposed final amendment

Reference.

Memorandum dated September 23, 1997, from Malcolm Knapp, Office of Nuclear Regulatory Research, to L. Joseph Callan, Executive Director for Operations, Subject: Final Amendments to 10 CFR Part 73, "Changes to Nuclear Power Plant Security Requirements"

cc. J Hoyle, SECY
J Blaha, OEDO
J. Mitchell, OEDO
M Knapp, RES
S Frattali, RES

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

December 10, 1997

Mr. L. Joseph Callan
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Mr. Callan:

SUBJECT PROPOSED FINAL REVISION 1 TO NUREG-1022, "EVENT REPORTING
GUIDELINES 10 CFR 50.72 AND 50.73"

During the 447th meeting of the Advisory Committee on Reactor Safeguards, December 3-6, 1997, we reviewed the proposed final Revision 1 to NUREG-1022, "Event Reporting Guidelines 10 CFR 50.72 and 50.73." We had the benefit of discussions with representatives of the NRC staff and of the documents referenced

The staff issued drafts of Revision 1 to NUREG-1022 for public comment in 1991 and 1994. Based on reconciliation of the public comments, the staff prepared a proposed final Revision 1 that combines the information contained in NUREG-1022 and its Supplements 1 and 2, the Statements of Considerations for 10 CFR 50.72 and 50.73, and other reporting guidance such as generic letters, bulletins, and information notices. The staff plans to issue Revision 1, move expeditiously to develop a rulemaking plan, and proceed with rulemaking regarding event-reporting requirements. The proposed amendment and revised event-reporting guidance are to be developed concurrently.

The proposed final Revision 1 to NUREG-1022 consolidates existing guidelines into a single reference document that provides a definitive baseline for event reporting. We recommend that the staff issue the proposed final Revision 1 to NUREG-1022.

We also recommend that the staff identify the items added to Revision 1 that the staff expects to include in the 10 CFR 50.72 and 50.73 rulemaking. The voluntary reporting in Revision 1 that the staff plans to make a requirement in a future amendment to 10 CFR 50.72 and 50.73 should be explicitly identified so that

licensees can plan for the implementation of these additional reporting requirements.

Sincerely,

A handwritten signature in black ink, appearing to read "R. L. Seale". The signature is fluid and cursive, with the first name "R." and last name "Seale" clearly distinguishable.

Robert L. Seale
Chairman

References:

1. Memorandum dated November 7, 1997, from Charles E. Rossi, Office for Analysis and Evaluation of Operational Data, NRC, to John T. Larkins, Executive Director, ACRS, Subject. Forwarding of NUREG-1022, Revision 1, "Event Reporting Guidelines, 10 CFR 50.72 and 50.73 "
2. Memorandum dated December 20, 1991, from Raymond F. Fraley, Executive Director, ACRS, to James M. Taylor, Executive Director for Operations, NRC, Subject Proposed Rulemaking to Modify Operating Power Reactor Event Reporting Requirements - 10 CFR 50.72 and 50.73.

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

December 12, 1997

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

Dear Chairman Jackson:

SUBJECT: PROPOSED REVISIONS TO 10 CFR 50.59 (CHANGES, TESTS AND EXPERIMENTS)

During the 447th meeting of the Advisory Committee on Reactor Safeguards, December 3-6, 1997, we met with representatives of the NRC staff and the Nuclear Energy Institute (NEI) to discuss proposed revisions to 10 CFR 50.59 (Changes, Tests and Experiments). We also discussed the proposed alternate rulemaking language proposed by NEI and guidance contained in NEI 96-07, Revision 0A, "Guidelines for 10 CFR 50.59 Safety Evaluations." We had the benefit of the documents referenced.

As a first step, the staff proposes a set of revisions to 10 CFR 50.59 that would clarify the current rule with respect to: (1) the criteria to be used to determine what constitutes an "unreviewed safety question" and (2) the language that requires "zero increase" in probability and consequences. The staff's stated intent is to continue developing a second-phase rule that would make the 10 CFR 50.59 process more risk-informed.

We support this two-step process because we agree with the staff and industry that there is an urgent need for stabilization of the 10 CFR 50.59 process. The proposed phase one revisions to the rule can provide interim stabilization. However, we believe the constraint of "zero increase" in the proposed revisions will serve to exacerbate the problem of excessive staff resources being required to review a large number of changes that are risk insignificant.

Therefore, in the second phase, we urge that the development of a new risk-informed rule be continued on an expeditious schedule. This rule should eliminate the "zero-increase" criteria and, instead, take as a starting point the

position that qualifying changes have effects on risk that are considered too small to require quantification of either the magnitude or the direction of change. Because probabilistic risk assessments (PRAs) will be insensitive to the types of changes made under 10 CFR 50.59, it will be challenging for the staff to develop a set of performance criteria for guidance to licensees for determining what changes qualify for consideration within the revised 10 CFR 50.59 process. It is essential that such performance criteria, rooted in the concepts of very small risk effects and compatibility with the proposed Regulatory Guide 1.174 (formerly DG-1061) process, be developed as guidance for implementing the risk-informed rule.

Sincerely,

A handwritten signature in black ink, appearing to read "R. L. Seale". The signature is fluid and cursive, with the first letters of the first and last names being capitalized and prominent.

R. L. Seale
Chairman

References:

- 1 Memorandum dated November 24, 1997, from Jack W. Roe, Office of Nuclear Reactor Regulation, NRC, to John T. Larkins, ACRS, transmitting summary information on 10 CFR 50.59 Rulemaking for December 4, 1997, meeting with ACRS.
- 2 U. S. Nuclear Regulatory Commission, Proposed Rule, 10 CFR Part 50 (and Part 60, 72, and 76), Changes, Tests and Experiments, dated November 6, 1997.
- 3 Nuclear Energy Institute, NEI 96-07, Draft Revision 0A, "Guidelines for 10 CFR 50.59 Safety Evaluations," July 1997.
- 4 Report dated October 9, 1997, from R. L. Seale, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: "Proposed Changes to 10 CFR 50.59 and Proposed Revision 1 to Generic Letter 91-18."
5. Report dated April 8, 1997, from R. L. Seale, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: "Proposed Regulatory Guidance Related to Implementation of 10 CFR 50.59 (Changes, Tests and Experiments)."

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

February 21, 1997

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

Dear Mr. President:

I am pleased to transmit to the Congress the 1996 report of the Advisory Committee on Reactor Safeguards on the U. S. Nuclear Regulatory Commission's Safety Research Program. This report is required by Section 29 of the Atomic Energy Act of 1954, as amended by Section 5 of Public Law 95-209.

Sincerely,

A handwritten signature in cursive script, reading "R. L. Seale", is positioned above the printed name.

R. L. Seale
Chairman

Enclosure:

U. S. Nuclear Regulatory Commission, "The Advisory Committee on Reactor Safeguards Report on Nuclear Safety Research and Regulatory Reform," dated February 1997

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

February 21, 1997

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

Dear Mr. Speaker:

I am pleased to transmit to the Congress the 1996 report of the Advisory Committee on Reactor Safeguards on the U. S. Nuclear Regulatory Commission's Safety Research Program. This report is required by Section 29 of the Atomic Energy Act of 1954, as amended by Section 5 of Public Law 95-209.

Sincerely,

A handwritten signature in cursive script, reading "R. L. Seale", is positioned above the printed name.

R. L. Seale
Chairman

Enclosure:

U. S. Nuclear Regulatory Commission, "The Advisory Committee on Reactor Safeguards Report on Nuclear Safety Research and Regulatory Reform," dated February 1997

**A REPORT TO THE CONGRESS
OF THE UNITED STATES OF AMERICA**

ON

NUCLEAR SAFETY RESEARCH AND REGULATORY REFORM

BY

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

FEBRUARY 1997

THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS REPORT
ON NUCLEAR SAFETY RESEARCH AND REGULATORY REFORM

The Advisory Committee on Reactor Safeguards, in the past, reported on very specific reactor safety research issues and programs. In light of the diminished resources available to support the U.S. Nuclear Regulatory Commission's Safety Research Program, we have chosen, instead, to report on the potential effects of a reduced Safety Research Program on regulatory reform and the ability to provide adequate safety oversight for a changing nuclear industry.

A vigorous research program dealing with the safety of commercial nuclear power production has served the Nuclear Regulatory Commission and the public well in the past. The continued availability of unbiased safety research information will be essential as the Nuclear Regulatory Commission establishes itself as the leader in the national effort to reform the regulatory process to focus on real risks, continued safety of operating nuclear power plants, and the performance of licensees. At the same time, initiatives taken by the commercial nuclear power industry in response to ongoing and anticipated deregulation of electrical power generation make it even more important that the Nuclear Regulatory Commission continue to have a Safety Research Program that provides the information needed to modify and improve its regulations to protect public health and safety.

From the inception of the civilian use of nuclear energy to generate electrical power, public safety has been of paramount concern. Initially, little experience and few industrial safety standards were available to ensure that nuclear power could be generated safely. As a result, prescriptive, highly conservative approaches that blanketed all aspects of nuclear power generation

were adopted by both the regulatory authority and the industry. Faults and vulnerabilities identified through operation of nuclear power plants were used to add layers of protection on this regulatory structure. Indeed, regulation of nuclear power generation has been successful in protecting public safety in this country. But, safety has been achieved through highly conservative regulation at great cost to both the producers and consumers of nuclear power.

As nuclear power generation has matured, experience has been gained in our understanding of the real risks of nuclear power. The Safety Research Program has enabled the Nuclear Regulatory Commission to develop a method called probabilistic risk assessment that can provide quantitative measures of these risks. The sophistication of this understanding has reached the point that it is now possible to initiate a reformation of the regulatory structure for nuclear power generation. This reformation will focus attention on what is significant to safety and at the same time will allow the industry to identify and use cost-effective strategies to mitigate risks. Reformation of regulation of all types to focus on risk is, of course, a national priority. The Nuclear Regulatory Commission is taking the lead in this national effort with its policy of risk-informed and performance-based regulation. Based on information that has come from the Safety Research Program, operational experience, and the ability to quantify risk, the Nuclear Regulatory Commission has been able to set forth safety goals that define how safe is safe enough. By working with individuals experienced in plant operations and using the tools of risk analysis, the NRC can now identify regulations that do not contribute to safety, and it will be able to define a rational, cost-benefit basis for imposing additional regulatory requirements.

Steps are being taken in the direction of risk-informed and performance-based regulation. The performance-based maintenance rule (10 CFR 50.65) is a tangible accomplishment. Rather than imposing bureaucratic prescriptions on every aspect of safety system maintenance, this rule allows the industry to find creative strategies to meet performance objectives approved by the Nuclear Regulatory Commission based on risk information. Satisfactory performance by licensees is rewarded by reductions in regulatory burdens while performance failures elicit increased regulatory scrutiny.

The Safety Research Program has aided the Nuclear Regulatory Commission in the development of standards for regulatory use of risk assessment. This would permit additional uses of this approach to focus dwindling resources on issues of most importance for protecting public health and safety. Target applications of these new standards are in-service inspection, in-service testing, and technical specifications for reactor safety systems. Continued research will be essential for further regulatory reforms.

New challenges to the regulation of nuclear power are emerging. These challenges come from the deregulation of electrical energy production and the need for the nuclear power industry to become more cost competitive. The nuclear industry is aggressively pursuing changes to remain economically viable. These changes could have significant safety implications that will require regulatory approval when they affect the licensing basis for nuclear power plants. Among the changes under consideration are increased fuel lifetimes, elevated operating power, digital instrumentation and control systems, and downsized work forces. Each of these changes could challenge the existing regulations for the protection of public health and safety. We believe that applied regulatory research programs will be required to develop bases/criteria for regulatory approval of these changes. Of

particular importance are the changes that may affect human performance in the operation of nuclear power plants.

Funding for research activities has fallen by a factor of about 3 over the last 10 years and all evidence points toward continued reductions in the future. While much of this decrease can be attributed to the maturation of the technology, funding for the Safety Research Program has been reduced to a level that may not allow a cost-effective response to these new challenges. The Nuclear Regulatory Commission now does not have the technical tools needed to evaluate all of the safety implications of extending fuel lifetimes to the extent the nuclear industry has requested. It cannot evaluate quantitatively the risk implications of personnel reductions and modernization that are being proposed by the nuclear industry. The Safety Research Program will have to be sustained and even augmented if the Nuclear Regulatory Commission is to complete its transformation to risk-informed and performance-based regulatory approach. Without the needed research support, the Nuclear Regulatory Commission may be forced to rely on historical, conservative, costly regulations not necessarily focused on risks. Safety innovations by the industry may be stifled. The opportunity to use regulation of nuclear power as an example of successful regulatory reform may be lost.

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

April 11, 1997

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

Dear Chairman Jackson:

SUBJECT: RISK-BASED REGULATORY ACCEPTANCE CRITERIA FOR PLANT-SPECIFIC APPLICATION OF SAFETY GOALS

In our December 6, 1996 meeting with the Commission, we committed to provide an example of how risk-acceptance criteria could be developed directly from the Safety Goals. Additionally, in a Staff Requirements Memorandum dated January 14, 1997, the Commission asked for our views on the relationship between the concept of "adequate protection," as used in the NRC regulations, and the NRC Safety Goals, from the standpoint of level of risk.

During the 440th meeting of the ACRS, April 3-4, 1997, we completed our deliberations on plant-specific application of NRC Safety Goals and the relationship between the concept of "adequate protection" and the Safety Goals. In our November 18, 1996 report on this subject, we stated that "the safety goals and subsidiary objectives can and should be used to derive guidelines for plant-specific applications." We noted that full-scope Level 3 probabilistic risk assessments (PRAs) would be necessary to use the quantitative health objectives (QHOs) directly to assess the acceptability of plant-specific risk. We also stated that this assessment of risk could be done in terms of the QHOs, along with the core damage frequency (CDF), or in terms of the CDF and large, early release frequency (LERF).

This report further discusses the need for plant-specific application of risk-acceptance criteria and the appropriateness of these criteria being derived from the Safety Goal QHO on early fatalities. The additional comments to this report provide examples of approaches that could be used to quantify lower tier acceptance criteria (i.e., LERF, or CDF and conditional containment failure probability) that will ensure that the early fatality QHO is met at each site. Quantification of the LERF at each site is needed to ensure the appropriateness of the choice of the LERF acceptance criterion proposed in draft Regulatory Guide DG-1061 and draft Standard Review Plan sections that support risk-informed, performance-based regulation.

Need for Plant-Specific Application

The Safety Goal Policy Statement makes it clear that the QHOs and the subsidiary goal on CDF were intended only to provide standards for the NRC to judge the overall effectiveness of its regulatory system. The Policy Statement specifically precludes enforcement of the Safety Goals on a plant-specific basis.

In the development of draft Regulatory Guide DG-1061 and the associated draft Standard Review Plan sections in support of risk-informed, performance-based regulation, the staff has found it necessary to propose risk-acceptance guidelines that can be applied on a plant-specific basis. These guidelines would be used, along with other considerations and inputs, for making judgments on the acceptability of requested changes to a licensee's current licensing basis. Reviewing plant-specific license amendments by using risk-acceptance guidelines is a positive action toward risk-informed, performance-based regulation.

We also note that, in the longer term, the Commission may want to consider having a quantified acceptable risk level to replace the current concept of "adequate protection." This risk level could eventually serve as an objective risk-acceptance criterion for many enforcement decisions.

Risk-Informed, Performance-Based Regulation

The Commission has directed the staff to increase the use of PRA in the regulatory process. We have endorsed this because we believe that a risk-informed, performance-based regulatory approach will lead to increased coherence in the regulatory system, to enhanced decision-making ability, and to technically defensible bases for granting regulatory relief.

A risk-informed, performance-based regulatory system ought not be implemented without the existence of top-level risk-acceptance criteria. The obvious choices for these criteria are the NRC Safety Goal QHOs. As it is the responsibility of the NRC to license individual plants and ensure adequate protection, there seems to be no alternative to plant-specific applications.

Relationship Between Adequate Protection and the Safety Goals

Currently, licensing acceptance criteria are embodied in the concept of "adequate protection." With this concept, a plant that is licensed and complies fully with the applicable rules and regulations, is considered to meet the "adequate protection" standard. "Adequate protection" embodies protection of public health and safety against threats that can be quantified in terms of risk as well as threats, such as sabotage and diversion of special nuclear material, for which the risk cannot now be quantified. In the discussion that follows, the nonquantifiable aspects of adequate protection are set aside. Since there are many ways in which plants can be designed and operated within the confines of the regulations, the natural result is a spectrum of risk levels across the population of operating plants. This conclusion is consistent with the results of the recent Individual Plant Examination Program. Since each licensed plant

must, by definition, provide adequate protection, the licensed plant that poses the highest level of risk places a bound on the quantified level of risk to be associated with "adequate protection."

Within the spectrum of risk, it is likely that there are plants with risk levels above the Safety Goals and other plants with risk levels below. If this is indeed the case, a single risk level that bounds "adequate protection" would be a risk level greater than the Safety Goal level. For those plants with risk levels below the Safety Goals, the difference between the plant risk and the Safety Goals can be viewed as margin. It is from some portion of this margin that plant-specific regulatory relief could be granted. For those plants with risk levels greater than the Safety Goals, the challenge will be to eventually reduce their risk to below the Safety Goal level within the confines of the backfit rule.

Regulatory Transparency

The unquantified "adequate protection" concept is not well understood by the general public because the public is unfamiliar with the regulatory process, the body of nuclear regulations, and associated underlying technical bases. We believe that a long-term objective of replacing the "adequate protection" concept with a well articulated and quantified "acceptable level of risk" if achievable, would enhance the public's understanding and acceptance of the regulatory process and would lead to a more uniform level of protection for all individuals living in the vicinity of nuclear plants.

We note that the use of risk-acceptance criteria such as the QHOs will add stability to the regulatory process. This is because the Safety Goals are determined primarily from considerations of societal risk, while the NRC rules and regulations, which are now used to specify adequate protection, change with time as our understanding of reactor safety issues evolves.

Safety Goals as Risk-Acceptance Criteria

It is our opinion that the QHOs are the appropriate choices for risk-acceptance criteria for plant-specific applications. The Safety Goals are the expression by NRC for "how safe is safe enough." In our opinion, this is what risk-acceptance criteria ought to be. As we stated in our August 15, 1996 report, the subsidiary CDF goal should be elevated to the status of a fundamental goal. Elevating the CDF subsidiary goal to the status of a fundamental goal can be considered as a defense-in-depth principle that provides balance between prevention and mitigation.

The early fatality QHO generally controls the risks from nuclear plant operations. Our understanding of risk associated with low-power and shutdown operations, or accidents initiated by external events in which emergency response

is impeded, is not yet sufficient to draw definitive conclusions concerning the limiting QHO in these situations.

Additional comments by ACRS Member T. S. Kress are presented below.

Sincerely,

R. L. Seale
Chairman

Additional Comments by ACRS Member T. S. Kress

While I agree completely with the Committee's report, I think it could be augmented in two respects. First, it could make it clearer that, with respect to plant-specific application of the Safety Goals, we are making two related, somewhat radical proposals — the second more so than the first:

- 1) That lower tier risk-acceptance criteria (CDF and LERF), now being proposed in Draft Regulatory Guide DG-1061 for use in making decisions regarding requested changes to a licensee's current licensing basis, be derived directly from the prompt fatality QHO and be of such value as to bound all current sites.
- 2) That, in the long run for enforcement purposes, the prompt fatality QHO be considered as the quantification of a risk level to replace "adequate protection."

Second, guidance on how lower tier criteria are to be derived from the QHO is needed. Consequently, I am including two attachments to these additional comments (one developed by me and a complementary one developed by ACRS Senior Fellow Rick Sherry). These provide examples of how to more rigorously derive the lower tier criteria. It is suggested that the staff consider these for use if the first proposal above is to be implemented.

Attachments:

1. Kress, T. S., "Risk-Based Regulatory Acceptance Criteria for Plant-Specific Application of Safety Goals," March 1997
2. Sherry, R. R., "Methodology for Estimating Offsite Early Fatality Risk in the Absence of a Level 3 PRA," March 1997

References:

1. Staff Requirements Memorandum dated January 14, 1997, from John C. Hoyle, Secretary, NRC, to John T. Larkins, Executive Director, ACRS, Subject: Meeting with ACRS, 9:30 A.M., Friday, December 6, 1996, Commissioners' Conference Room.

2. Report dated November 18, 1996, from T. S. Kress, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: Plant-Specific Application of Safety Goals.
3. Report dated August 15, 1996, from T. S. Kress, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: Risk-Informed, Performance-Based Regulation and Related Matters.
4. U.S. Nuclear Regulatory Commission, NUREG-1560, Volume 1, Part 1, "Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance," Summary Report, Draft Report for Comment, October 1996.
5. U.S. Nuclear Regulatory Commission Draft Regulatory Guide, Draft DG-1061, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Current Licensing Basis," dated February 28, 1997 (Predecisional).
6. U.S. Nuclear Regulatory Commission, Draft Standard Review Plan Chapter 19, Revision L, "Use of Probabilistic Risk Assessment in Plant-Specific, Risk-Informed Decisionmaking: General Guidance," dated March 3, 1997 (Predecisional).

ATTACHMENT 1

Risk-Based Regulatory Acceptance Criteria for Site Specific Application of Safety Goals

T.S. Kress

The purpose of this discussion paper is to explore the concept of using the Safety Goal quantitative health effects (QHO) on early fatalities to derive lower tier risk acceptance criteria for application on a plant-specific basis. A starting point for expressing the early fatality QHO in a form that can be used to derive different tier criteria is the following working definition for the risk of early fatalities for any specific plant in terms of the normal determinations of probabilistic risk assessments (PRAs).

$$\text{Mean number of early fatalities} \equiv \sum_i (CDF)_{i,k} (CCFP)_{i,k} (C_{ef})_{i,k} \quad (1)$$

where k refers to a specific plant,
 i refers to the spectrum of accident sequences,
 $(CDF)_{i,k}$ is the core damage frequency for sequence i of plant k ,
 $(CCFP)_{i,k}$ is the conditional containment failure probability for sequence i and plant k ,
 $(C_{ef})_{i,k}$ is the early fatality consequences at site k given the sequence i which has associated with it a source term St_i , that may be defined in terms of the equivalent release of iodine *to the outside environment*.

The QHO objective for early fatalities is expressed in terms of individual risk. The Safety Goal Policy Statement specifically states that the early fatality QHO is to be determined by calculating the cumulative individual fatalities *within one mile* of the site boundary, C_{ef1} , and dividing that by the population within that same one mile region, P_{k1} . Therefore, Equation 1, for purposes of comparing with the early fatality QHO, should be rewritten as

$$(IR) \text{ Individual risk} \equiv \sum_i (CDF)_{i,k} (CCFP)_{i,k} (C_{ef1})_{i,k} / P_{k1} \quad (2)$$

In order to proceed further, we first note that, in general, the early fatality consequences within one mile of the site boundary can be related to the source term expressed in terms of the equivalent release of iodine by the relationship

$$C_{ef1} \sim (St)^n$$

An appropriate exponent is $n = 0.9$, which can be obtained from Figure 2 of Attachment 2. Consequently, if a calculation were available that gave the expected early fatalities within one mile of the site boundary, C_{k1r} , for any reference source term, St_r , then Equation 2 would be rewritten as

$$IR \equiv \sum_i (CDF)_{i,k} (CCFP)_{i,k} \left[\frac{C_{klr}}{P_{kl}} \left(\frac{St_{ik}}{St_r} \right)^n \right] \quad (3)$$

For our present purposes, Equation 3 can be rewritten as

$$\sum_i (CDF)_{i,k} (CCFP)_{i,k} (St_{i,k})^n \equiv \frac{(IR) P_{kl} (St_r)^n}{C_{klr}} \quad (4)$$

The items on the left of Equation 4 are those that are determined by a full-scope Level 2 PRA with source term (to the environment) capability. The items on the right contain the result of a Level 3 consequence analysis for IR and a site characterization parameter, $(P_{kl})(St_r)^n/C_{klr}$. This parameter can easily be determined using an appropriate "consequence" code such as MACCS or CRAC and, in fact, may already exist as part of NUREG/CR-2239, "Technical Guidance for Siting Criteria Development."

NUREG/CR-2239 basically has a characterization of each US site with respect to C_{et} . The CRAC code was used to determine site-specific early fatality consequences for

- a standard 1120 Mwe plant (for inventory),
- summary evacuation assumptions,
- actual site population and wind rose,
- best-estimate meteorology, and
- a variety of source terms.

Although the values for P_{kl} and C_{klr} out to the one mile boundary were not specifically reported in NUREG/CR-2239, the information may still exist in the archival print-outs of the computed output. In case these data are not currently available, Attachment 2 presents a convenient and robust method for estimating the prompt fatalities at each site for any reference source term and provides an appropriate definition of LERF.

Direct Comparison with the QHO Using Level 2 PRAs with Source Term Capability

Equation 4 is an abbreviated form of the early fatality IR that can be used to derive an acceptance criterion for plants with full-scope, Level 2 PRAs with source term capability. For the QHO criterion to be met, the IR must be $\leq 5 \times 10^{-7}/\text{yr}$. Therefore the acceptance criterion parameter is clearly

$$\sum_i (CDF)_{i,k} (CCFP)_{i,k} (St_{i,k})^n \leq \frac{(5 \times 10^{-7}) P_{kl} (St_r)^n}{C_{klr}} \quad (5)$$

The criterion, Equation 5, can also be cast in terms of a conditional early release probability criterion, CERP, that can be defined as

$$[\sum (CDF)_{i,k}] \times CERP \equiv \sum (CDF)_{i,k} (CCFP)_{i,k} (St_{i,k})^n$$

or

$$CERP_k \equiv \frac{\sum_i (CDF)_{i,k} (CCFP)_{i,k} (St_{i,k})^n}{\sum_i (CDF)_{i,k}}$$

In terms of acceptance criteria similar to what was recently proposed by the staff with coordinates of CDF and LERF with a basic tenet that CDF should be $\leq 1 \times 10^{-4}/\text{yr}$, the appropriate plant specific CERP acceptance criterion would be

$$CERP_{ka} \leq \frac{(5 \times 10^{-7})(St_r)^n (P_{kl})}{(10^{-4})C_{klr}}$$

If a single $CERP_a$ is desired that would bound all current sites, the site with minimum P_{kl}/C_{klr} would have to be chosen.

Level 2 PRA Without Source Term Capability

For the case of a plant that has Level 2 PRA capability to determine CCFP but not to establish the various associated sequence source terms, it is desired to develop a lower tier criterion in terms of a defined CCFP that would be bounding with respect to ensuring that the early fatality QHO is met on a plant-specific basis.

To do this from Equation 5, it will be necessary to use a single source term that is sure to bound each sequence source term. A possible value that would be bounding (for PWRs only) is 0.5 (release fraction of iodine equivalent). With this value, Equation 5 as a PWR plant-specific bounding criterion becomes

$$\sum_i (CDF)_{i,k} (CCFP)_{i,k} \leq \frac{(5 \times 10^{-7})(P_{kl})(St_r)^n}{(C_{klr}) (.5)^n}$$

The parameter on the left is what I believe the staff used in its decision chart and called a "large early release frequency" (LERF). This would more aptly be called a containment failure frequency.

To express this in terms of a conditional containment failure probability, we could use the usual definition of CCFP.

$$CCFP_k \equiv \frac{\sum_i (CDF)_{i,k} (CCFP)_{i,k}}{\sum_i (CDF)_{i,k}}$$

Then, an acceptance value (for PWRs only) for use in a "decision" chart along with CDF = $10^{-4}/\text{yr}$ becomes

$$CCFP_{ka} \leq \frac{(5 \times 10^{-7})}{(10^{-4})} \frac{(P_{kl})}{(C_{klr})} \frac{(St_r)^n}{(.5)^n}$$

A separate acceptance criterion could be developed for BWRs using a bounding St that would have to recognize the effects of the suppression pool and its bypass. An optional way to account for the differences between BWRs and PWRs could be to make different choices for the sequences to go into calculating the CCFP. This is what the staff chose to do - thereby preserving the same acceptance criterion for both reactor types.

For site bounding values, the minimum P_{kl}/C_{klr} for each reactor type would be chosen. It would be this bounding value that should, in the case of PWRs, be compared with the $CCFP = 0.1$ that is often mentioned as an appropriate criterion to use along with $CDF = 1 \times 10^{-4} / \text{yr}$.

ATTACHMENT 2

Methodology for Estimating Offsite Early Fatality Risk in the Absence of a Level 3 PRA

Rick Sherry- Senior Fellow
Advisory Committee on Reactor Safeguards

Introduction

This paper defines a simple relationship between the Safety Goal Quantitative Health Objective (QHO) for Individual Early Fatality Frequency (IEFF) and a Large, Early Release Frequency (LERF) that can be used to estimate the Safety Goal QHO for a specific plant at a specific site. This paper also provides a quantitative definition of the LERF. The relationship utilizes simple site-specific characteristics (wind direction frequencies and population demographics) and results from a Level 2 plant-specific probabilistic risk assessment (PRA) (release category frequencies and source term characteristics).

Estimates using this methodology have been compared with results from detailed calculations performed using the MELCOR Accident Consequence Code System (MACCS) (Ref. 1) for the NUREG-1150 study (Ref. 2), the Grand Gulf Shutdown study (Ref. 3) and the Maine Yankee PRA/IPE. This comparison includes internally initiated sequences, seismic sequences, and shutdown sequences.

Summary of Methodology

The relationship between the IEFF Safety Goal QHO and the LERF is defined as:

$$IEFF = LERF \times EI \quad \text{Equation 1}$$

$$\text{where: } EI = \text{Exposure Index} = \frac{\sum_{i=1}^{16} P_i \times F_i}{\sum_{i=1}^{16} P_i} \quad \text{Equation 2}$$

F_i = the relative frequency wind blows toward sector i

P_i = population in sector i within one mile of the plant

$$\text{and: } LERF = \sum_{k=1}^{N_{rc}} RC_k (\text{: early } I > 10\% \text{ and Evacuation Delayed}) \times (1 - F_{no-evac}) \\ + \sum_{k=1}^{N_{rc}} RC_k (\text{: total } I > 10\%) \times F_{no-evac} \quad \text{Equation 3}$$

RC_k = the frequency of release category k

N_{rc} = number of release categories

$F_{no-evac}$ = the population fraction not evacuating

Discussion

The Exposure Index (EI) provides a measure of the average probability that a specific individual within one mile of the plant would be exposed to a lethal radiation dose (given that a release occurs from the plant of sufficient magnitude to produce lethal doses) and assuming that the individual does not evacuate. The exposure index defined above is a slight variation from that initially developed for the Generic Environmental Impact Statement for License Renewal of Nuclear Plants (Ref. 4). Table 1 shows the calculated values for the exposure index for a number of nuclear plant sites using site specific wind direction frequency and population data. Note that all calculated values of the EI fall well within a factor of two of the nominal value of 1/16 which would result if the population within one mile of the plant and the wind direction frequency were uniformly distributed in the sixteen wind direction sectors.

In the above definitions, the release categories which are included in the summations are either 1) those with an early release fraction of iodine greater than 10% of the core inventory and for which evacuation is delayed or 2) release categories where the total iodine release fraction (sum of early and later releases) exceeds 10% and evacuation does not occur. The second summation accounts for the fraction of individuals that do not evacuate and are assumed to be exposed to all time phases of the radionuclide release. This second summation also governs severe seismic sequences where evacuation is assumed to be ineffective.

A ten percent iodine release fraction has been selected as the threshold for source term magnitudes which can result in lethal doses. The selection of 10% was based principally on the work of Kaiser (Ref. 5) which indicated that a threshold for early fatalities occurs at a release fraction of approximately 10% of volatile fission product (I, Cs, Te) release fraction. Figure 1 from Kaiser illustrates this behavior.

An early release is defined as the release which occurs at the time of containment failure (assuming core damage has occurred prior to containment failure). For sequences where containment integrity has been lost prior to core damage, early release begins when core damage

commences. This definition for early release is identical to that provided in NUREG-1150. Typical periods of release duration for early release in the NUREG-1150 study are from several minutes to several hours.

Regardless of the magnitude of the source term, if evacuation commences sufficiently prior to the time when the release of radionuclides begins then the probability of early fatalities is dramatically reduced. In the NUREG-1150 study, three release category timing subgroups were defined for each release category. For subgroup 1, it was assumed that evacuation commenced at least 30 minutes prior to the start of radionuclide release. For subgroup 3, the start of evacuation was delayed until one hour or more after radionuclide release had begun. For subgroup 2, the start of evacuation was assumed to occur within a time window from 30 minutes before, to one hour after, the start of release.

Figure 2 illustrates the impact of these various evacuation assumptions on the Conditional Individual Early Fatality Probability¹ (CIEFP). This figure plots the CIEFP against the iodine release fraction. Individual data points for the three release category subgroups are shown with different symbols. This figure illustrates the effectiveness of early evacuation in reducing the Individual Early Fatality Risk. The diamond shaped symbols represent sequences for which evacuation was delayed until one hour or later after the start of radionuclide release (subgroup 3). For these sequences, the peak CIEFP is on the order of $3 \times (10)^{-2}$. The triangle shaped symbols represent sequences for which evacuation was initiated at least 30 minutes prior to the start of the release of radionuclides (subgroup 1). These sequences have a CIEFP of order $2 \times (10)^{-4}$ or less. These results are dominated by the fraction of the affected population who are assumed not to evacuate. For sequences characterized by evacuation commencing at about the same time as the start of radionuclide release (subgroup 2), the results (shown as circles) generally fall between the results for subgroups 1 and 3.

Equation 1 can be rearranged and used to estimate a plant specific LERF subsidiary objective:

$$LERF = \frac{IEFF}{EI} \quad \text{Equation 4}$$

This equation along with the plant specific Exposure Index values shown in Table 1 have been used to calculate a plant specific LERF objective. These results are also shown in Table 1. Note

¹ The CIEFP is conditional on the occurrence of sequences (release categories) with the indicated volatile fission product release magnitude.

that for these calculations, the assumed value for the Safety Goal QHO IEFF is taken as $3 \times (10)^{-7}$ per year.²

Example Application Results

The simple methodology has been applied to six cases. These cases are:

- Surry Internal Event Initiated Sequences (NUREG-1150)
- Surry Seismic Sequences (NUREG-1150)
- Peach Bottom Internal Event Initiated Sequences (NUREG-1150)
- Peach Bottom Seismic Sequences (NUREG-1150)
- Grand Gulf Shutdown Sequences (NUREG/CR-6143)(Ref. 3)
- Maine Yankee Internally Initiated Sequences (PRA/IPE submittal)

The results of the example application are summarized in Figure 3. In this figure, the Individual Early Fatality Risk Frequency calculated using the simple methodology is compared with the mean and 95th percentile values from the PRA offsite consequence (MACCS) code calculations. In all cases, the simple methodology produces estimates for the IEFF that are above the PRA mean values. For the two seismic cases, the results fall between the PRA mean and 95th percentile values, and in the remaining cases the results lie above the 95th percentile (but within a factor of three of the 95th percentile value).

The limited comparison summarized in Figure 3 indicates that this methodology may provide a simple and easy to use approach for providing reasonably robust estimates for the IEFF for PRA analyses lacking a detailed Level 3 offsite consequence. This methodology has been applied to a broad spectrum of PRA sequence classes and in all cases the comparison with the PRA results have been favorable.

Discussion of Results

Internally Initiated Sequences (NUREG-1150)

For Peach Bottom internally initiated sequences, all release categories that met the criteria for early iodine release greater than 10% also had evacuation beginning prior to the start of release. This is the principal reason that the MACCS calculated IEFF is low for Peach Bottom. It is only the small fraction of the population that does not evacuate which contributes to the calculated

² Based on data from the National Center for Health Statistics the total U.S. societal accident fatality rate was 34.4 and 34.6 deaths per 100,000 individuals in 1993 and 1994, respectively. This translates into an individual early fatality Safety Goal QHO of $3 \times (10)^{-7}$ per reactor year.

IEFF. Consequently, in this study the estimated IEFF for Peach Bottom is controlled by the second summation term in Equation 3 (as for the NUREG-1150 MACCS analyses it was assumed that 0.5% of the population did not evacuate and $F_{\text{no-evac}}$ was set to this value).

For the Surry plant, the dominant contributors to a large early fission product release and early fatalities for internally initiated sequences were containment bypass events (12% of CDF). These sequences contribute about 80% of the early fatality risk. For the fast developing interfacing system LOCA bypass sequences, warning times were typically within one hour of the start of radionuclide release. Consequently, evacuation would generally begin after the start of release.

Seismic Sequences (NUREG-1150)

The NUREG-1150 results for seismic sequences for the Peach Bottom and Surry plants using the LLNL seismic hazard curves were compared for this study. The NUREG-1150 seismic sequence consequence analyses were separated into two categories - low Peak Ground Acceleration (PGA) (< 0.6 g) and high PGA (> 0.6 g). For seismic sequences, the offsite emergency protective action assumptions input into the MACCS analyses differed from that for internally initiated events. Particularly important was the assumption that for the high PGA category sequences evacuation was assumed to be ineffective (i.e., does not occur). Consequently, for this study the total iodine release fraction for high PGA seismic sequence classes was used (i.e. the second summation in Equation 3 with $F_{\text{no-evac}}$ set to one).

Grand Gulf Shutdown Sequences (NUREG/CR-6143)

For shutdown sequences, the activity level of important volatile fission product species will be reduced by decay from their full power levels and this reduction should be accounted for in analyses. However, for this application this was not done. Even with this limitation the estimated IEFF using the simple methodology was in good agreement with the MACCS results. As for the Peach Bottom internally initiated sequences, warning times for Grand Gulf shutdown sequences were sufficiently early that evacuation commenced prior to the beginning of fission product release, and it is only that small fraction of the population that does not evacuate which contributes to the calculated IEFF. Consequently, the estimated IEFF is controlled by the second summation term in Equation 3.

Maine Yankee Internally Initiated Sequences (PRA/IPE)

The IEFF for Maine Yankee is dominated by sequences where containment failure occurs as a result of a hydrogen combustion event near the time of reactor vessel meltdown. This type of failure was predicted to occur for about 8% of the core damage frequency. For those sequences contributing to the IEFF, the radionuclide release is predicted to begin about one hour after a general emergency would be expected to be declared. In the Maine Yankee PRA, delay times

ranging from 1 to 2.5 hours were assumed (dependent on the time of year). Hence, for these sequences, radionuclide release generally begins prior to, or coincident with, the start of evacuation.

Enhancements - Use of Iodine Equivalent

Although this paper has shown that reasonably good correlation exists between the IEFF and the LERF calculated using Equation 3 (which uses an iodine release fraction of 10% as threshold for early fatalities), it was also observed that under certain conditions other fission product species groups can make significant contributions to the early fatality risk. It was observed that if the Ru or La group release fractions exceeded about 1 to 5% of core inventory, they began to make a significant contribution to early fatality risk. Furthermore, as discussed above, for sequences initiated during shutdown which may have significant decay periods, the contribution of iodine isotopes to the early fatality risk will be significantly reduced and other radionuclide groups will begin to dominate. For these reasons, a parameter that can be evaluated from the release fraction for all radionuclide groups (and as a function of shutdown time) is desirable. An example of the type of iodine equivalent parameter that is required is presented in NUREG/CR-5164 (Ref. 6) which is based on work reported in Reference 7. In these studies, weighting factors for the contribution to early fatality risk for each radionuclide species group were developed. These weighting factors (which are multiplied by the radionuclide group release fraction) can be used to develop a total iodine equivalent release.

References

1. Chanin, D. I., et al., "MELCOR Accident Consequence Code System (MACCS): User's Guide," NUREG/CR-4691, Vol. 1, Sandia National Laboratories, February 1990.
2. "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," NUREG-1150, December 1990. [Note, references to NUREG-1150 also imply reference to the NUREG-1150 supporting documents including applicable volumes of NUREG/CR-4551]
3. Brown T. D. et al., "Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Grand Gulf, Unit 1," NUREG/CR-6143, Vol. 6, Part 1, March 1995.
4. "Generic Environmental Impact Statement for License Renewal of Nuclear Plants", NUREG-1437, Vol. 1, May 1996.
5. G.D. Kaiser, "The Implications of Reduced Source Terms for Ex-Plant Consequence Modeling," ANS Executive Conference on the Ramifications of the Source Term, Charleston, SC, March 12, 1985.
6. Madni I. K., et al., "A Simplified Model for Calculating Early Offsite Consequences from

Nuclear Reactor Accidents'" NUREG/CR-5164, Brookhaven National Laboratory, July 1988.

7. Alpert J., et al, "Relative Importance of Individual Elements to Reactor Accident Consequences Assuming Equal Release Fractions," NUREG/CR-4467, Sandia National Laboratories, March 1986.
5. Brown T. D. et al., "Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Grand Gulf, Unit 1," NUREG/CR-6143, Vol. 6, Part 1, March 1995.

Table 1
Example of Plant Specific Exposure Index and
Derived Subsidiary LERF Objective

Plant	EI	Derived Subsidiary LERF Objective
Grand Gulf	.065	5E-6
Surry	.077	4E-6
Sequoyah	.045	7E-6
Maine Yankee	.040	8E-6
Zion	.081	4E-6
Peach Bottom	.075	4E-6
Lasalle	.083	4E-6
Nominal	.063 (1/16)	5E-6

Figure 1. Early Fatalities

Figure 2
Surry Individual Early Fatality Risk
Internal Initiators

Figure 3
Individual Early Fatality Risk
PRA (NUREG-1150) v Simple Estimator

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

September 19, 1997

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

Dear Chairman Jackson:

SUBJECT: SITE-TO-SITE VARIATION IN RISK-BASED REGULATORY
ACCEPTANCE CRITERIA FOR PLANT-SPECIFIC APPLICATION OF
SAFETY GOALS

In the Staff Requirements Memorandum dated May 27, 1997, the Commission requested that the ACRS determine the change in core damage frequency (CDF) and large, early release frequency (LERF) from site-to-site when these lower-tier criteria are derived from the individual early fatality quantitative health objective (QHO). In response to this Commission request, during the 443rd and 444th meetings of the Advisory Committee on Reactor Safeguards, July 9-11 and September 3-5, 1997, we discussed the plant-specific application of NRC Safety Goals and derivation of subsidiary criteria. These criteria would be used in determining the acceptability of proposed changes to the licensing basis. During the discussions, we had the benefit of the documents referenced.

This report discusses the site variability in LERF as a risk-acceptance criterion derived from the individual early fatality QHO. The bases for the conclusions and recommendations in this report are provided in the attached studies. We addressed the CDF criterion in our April 11, 1997 report.

Variability in LERF Criteria Derived from the Safety Goal
Individual Early Fatality QHO

In support of preparing our response to the Commission's request, an ACRS Senior Fellow performed a study (Attachment 1) to answer the following questions:

- Is there sufficient site-to-site variability in the site characteristics important to individual early fatality risk to warrant site-specific determination of lower level acceptance criteria - e.g., LERF?
- Can this range of variability be evaluated and bounded?

- Can generic criteria or site-specific criteria be determined using simplified approximate methods?

The range of variability in individual early fatality risk due to the site-to-site variations in the parameters important to individual early fatality risk, such as site-to-site population distribution, wind direction frequency distribution, exclusion zone size, and meteorology record, was evaluated for all U.S. plant sites and was found to be relatively small (a variation of a factor of 4).

This study has been independently reviewed, and although the reviewers had different opinions on some of the details of the analysis, all of the reviewers concurred with the overall conclusion on the magnitude of the variability. Since this variability is much less than the magnitude of uncertainties associated with the probabilistic risk assessment (PRA) calculation of the LERF, this study concluded that the site-to-site variability in individual early fatality risk is insufficient to warrant development of site-specific LERF criteria. Hence, a single LERF criterion can be determined on a generic basis. This is consistent with the approach used by the staff in the draft Regulatory Guide DG-1061, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Current Licensing Basis."

We believe that the information provided in the study can be used to bound the variability of site-specific LERFs.

Adequacy of Individual Risk Metric

In addition to the individual risk metric, DG-1061 contains deterministic considerations that include other risk parameters — one of which is "siting factors." A second study, which was performed by an ACRS Senior Fellow (Attachment 2), noted that one such siting factor, site population density, is a robust indicator of total (societal) early fatality risk. Consequently, we recommend that the consideration of siting factors, mentioned in DG-1061 only in passing, be given much greater visibility and prominence as part of the decision making process.

Conclusions and Recommendations

We have determined that there is insufficient site-to-site variability in the factors that influence individual early fatality risk to warrant site-specific differences in the LERF subsidiary criterion.

Large site-to-site variations in the population density result in large variations in total early fatality risk. We recommend that this robust indicator of societal risk be made more explicit and

prominent in the criteria to be used in assessing plant-specific changes to the current licensing basis.

Sincerely,

R. L. Seale
Chairman

References:

1. Memorandum dated May 27, 1997, from John C. Hoyle, Secretary, NRC, to John T. Larkins, Executive Director, ACRS, Subject: Staff Requirements - Meeting with the ACRS, May 2, 1997, Commissioners' Conference Room.
2. Report dated November 18, 1996, from T. S. Kress, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: Plant-Specific Application of Safety Goals.
3. Report dated April 11, 1997, from R. L. Seale, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: Risk-Based Regulatory Acceptance Criteria for Plant Specific Application of Safety Goals.
4. U. S. Nuclear Regulatory Commission, NUREG/CR-2239, "Technical Guidance for Siting Criteria Development," Prepared by Sandia National Laboratories, December 1982.

Attachments:

1. Memorandum dated June 27, 1997, from R. Sherry, Senior ACRS Fellow to ACRS Members, Subject: Considerations for Plant-Specific, Site-Specific Application of Safety Goals and Definition of Subsidiary Criteria.
2. Memorandum dated June 11, 1997, from R. Sherry, Senior ACRS Fellow to ACRS Members, Subject: Consideration of Societal Risk in Plant-Specific, Site-Specific Application of Safety Goals and Definition of Subsidiary Criteria.

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

June 27, 1997

MEMORANDUM TO: ACRS Members

FROM: Rick Sherry, Senior ACRS Fellow

SUBJECT: CONSIDERATIONS FOR PLANT-SPECIFIC, SITE-SPECIFIC
APPLICATION OF SAFETY GOALS AND DEFINITION OF
SUBSIDIARY CRITERIA

Background

In its November 18, 1996 report (ACRS-1996), the ACRS stated that "the safety goals and subsidiary objectives can and should be used to derive guidelines for plant-specific applications." In its April 11, 1997 report (ACRS-1997), the ACRS stated further that "Quantification of the LERF [large, early release frequency] at each site is needed to ensure the appropriateness of the choice of the LERF acceptance criterion proposed in draft Regulatory Guide DG-1061 and draft Standard Review Plan sections that support risk-informed, performance-based regulation."

In a Staff Requirements Memorandum dated April 15, 1997 (SRM-1997a), the Commission stated (referring to *Direction Setting Issue 12*), "The staff should develop objective standard(s) for the application of risk-informed, performance-based and risk-informed less prescriptive approaches to regulations on an expedited basis. Such standard(s) could be in the form of individual plant safety goals and subsidiary objective performance criteria as discussed in the issue paper."

In a staff requirements memorandum dated May 27, 1997 (SRM-1997b), the Commission "requested that the ACRS determine the change in the CDF and LERF from site to site, when these lower-tier criteria are derived from the prompt fatality quantitative health objectives." This memorandum addresses this request from the Commission.

Issues

To respond to this request, the following issues need to be addressed:

- Is there sufficient site-to-site variability to warrant site-specific determination of lower-level acceptance criteria — e.g., LERF criteria?

ATTACHMENT 1

- Can this range of variability be evaluated and can an appropriate bound be established on this variability?
- Can generic criteria or site-specific criteria (if required) be determined using simplified approximate methods?

This memorandum will address these issues.

Parameters Important to Early Fatality Risk

This section discusses those parameters that are (potentially) important to early fatality risk. These will be discussed in two categories. The first category includes those parameters that are determined by plant design/plant operations and that should be captured in a proper definition of the LERF. The second category includes those parameters that are determined by site characteristics.

The plant design/plant operations related parameters potentially important to early fatality risk:

- Source Term Characteristics
 - magnitude of the fission product release from containment (particularly the volatile I and Te groups)
 - release thermal energy and release height
- Timing Characteristics
 - timing of release (relative to the start of protective actions) — effective evacuation begins before the start of radionuclide release.
 - absolute time of release relative to reactor shutdown (for radionuclide decay considerations)

The site-related parameters potentially important to the early fatality risk are:

- sector population distribution
- wind direction frequency distribution
- variability due to site-to site variations in local meteorology
- size of exclusion boundary

This memorandum focuses on this second category of parameters.

Simplified Model for Individual Early Fatality Risk

A simple relationship between the individual early fatality frequency (IEFF) and the LERF has been defined (Sherry-1997) as:

$$IEFF = LERF \times EI \quad \text{Equation 1}$$

$$EI = \text{Exposure Index} = \frac{\sum_{i=1}^{16} P_i \times F_i}{\sum_{i=1}^{16} P_i} \quad \text{Equation 2}$$

where: F_i = the relative frequency wind blows toward sector i
 P_i = population in sector i within one mile of the plant

For site-specific analyses, it has been our practice to consider the EI as a site-specific constant. In this analysis, the EI is treated as a random variable, which represents the variability (across the spectrum of sites) of the site-related parameters important to early fatality risk. The EI can be represented as:

$$EI \equiv f(p, w, r, m)$$

where: p = population distribution
 w = the wind direction frequency distribution
 r = minimum radius of the exclusion zone boundary
 m = the local meteorology

In the following analyses, we will construct the distribution for the EI. This will involve a three step process. First, we will determine the distribution resulting from site-to-site population and wind direction frequency variations. In the second step, we will add the variability from differences in exclusion zone boundary distances. Finally, we will add the variability from differences in the local meteorology.

Step 1: Variability in Population Distribution and Wind Direction Frequency

The exposure index for each nuclear plant site has been evaluated using the relationship shown in Equation 2. The wind direction frequencies for each plant site were taken from Appendix A of the Sandia Siting Study (NRC-1982) and are shown in Attachment 2. The

population distribution around each plant was calculated using the SECPOP90 computer code.¹ The SECPOP90 code contains a database of 1990/1992 U.S. Census data. The SECPOP90 code requires the map coordinates of each plant; these were obtained from the geospatial data available on the NRC Wide World Web server. These coordinates are shown in Attachment 1.

The safety goal policy statement notes that the QHO for individual early fatalities should be evaluated for the "average individual ... locationally who resides within a mile of the plant site boundary." In NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," the IEFF safety goal QHO was evaluated in the 1 mile annular ring with interior radius equal to the minimum distance to the (irregularly) shaped exclusion zone boundary. This interpretation of the evaluation region has been accepted for use in this study. Attachment 3 shows the minimum exclusion distances for each plant site (NRC-1982).

To simplify the analyses, the SECPOP90 code was set up to give the population distribution for three fixed radial rings (0-1 mile, 1-1.5 miles and 1.5-2 miles). Since the minimum exclusion radius of all but three plants is less than 1 mile, this range bracketed the evaluation distance (exclusion zone boundary + 1 mile) for most plants. For those plants with exclusion zones less than 1 mile, EIs were calculated for distances of 1, 1.5, and 2 miles from the plant and the EI at the exclusion zone boundary plus 1 mile was determined by interpolation. For the plants with exclusion zones greater than 1 mile, the EIs were calculated directly using the SECPOP90 code.

Table 1 shows the calculated EIs following step 1 for each plant. Figure 1 shows a histogram of the results. The statistics of the distribution are shown in Table 2. The mean and median values of the distribution are both approximately 0.61 and the 95th percentile value is 0.85, resulting in a ratio of the 95th percentile value to the mean value equal to 1.4.

Step 2: Variability Resulting From Differences in Site Boundary Radius

The radii of the minimum exclusion zone boundary range from a low of 0.17 mile to 1.33 miles, with a median value of 0.51 mile and a mean value of 0.56 miles. This range of differences potentially could have a significant effect on the site-to-site variability in individual early fatality frequency. In the Sandia siting study, a factor of $r^{-1.5}$ was used to represent the approximate decrease in risk with distance from the plant. Hence, for the range in exclusion zone boundary distances given above the potential site-to-site variation expressed as a risk ratio could be as much as:

In the above calculation the ratio is evaluated at the midpoint of the radial ring bounded on

¹ The potential for significant error in the population distribution close to the site exists due to the procedure for assigning population from the Census Block Data to the compass sectors and radial rings surrounding a plant. This potential uncertainty was not evaluated in this study.

$$\text{Risk Ratio} = \frac{\frac{1}{(0.17+0.5)^{1.5}}}{\frac{1}{(1.33+0.5)^{1.5}}} = 4.5$$

the inside by the exclusion zone minimum radius and on the outside by the exclusion zone minimum radius plus 1 mile. The potential variability due to differences in the exclusion zone boundary was evaluated by modifying the EI so that:

$$EI_r = \frac{\sum_{i=1}^{16} P_i \times F_i}{\sum_{i=1}^{16} P_i} \times \frac{1}{[1+(r-r_{med})]^{1.5}} \quad \text{Equation 3}$$

where: r = the site-specific exclusion zone minimum radius
 r_{med} = the median value of exclusion zone minimum radii

The formulation of the second term in this equation is such that at the median value for the exclusion radii this term is equal to 1. For values of r less than the median value the term is greater than 1 and for values of r greater than the median radius, the value is less than 1.

The calculated values for the exposure index shown in equation 3 for each site are provided in Table 1 and a histogram of the results is shown in Figure 2. The statistics of the distribution are shown in Table 2. The distribution has a mean value of 0.062, a median value of 0.058, and a 95th percentile value of 0.114. The ratio of the 95th percentile value to the mean value is 1.8.

Step 3: Variability Resulting From Differences in Local Site Meteorology

In addition to the variability in early fatality risk resulting from differences in site population distribution, wind direction frequency, and size of the exclusion zone, the local meteorology also contributes to site-to-site variability. In the Sandia siting study a sensitivity analysis was performed to determine the site-to-site variability potentially arising from differences in the local meteorology. This analysis found that there was less than a factor of two difference in the mean number of early fatalities over a set of 29 weather records selected to represent the spectrum of meteorological conditions found within the continental United States. The analysis results were derived from CRAC code calculations using a defined source term, the population distribution, and wind direction frequency for a high population density site. In these calculations, only the weather record was modified from calculation to calculation.

A distribution of individual early fatalities was constructed to represent the potential

variability in local meteorology and this distribution was then combined with the EI distribution determined from equation 2. The modified exposure index is shown in Equation 4.

$$EI_i = EI_r \times \epsilon_m \quad \text{Equation 4}$$

where: ϵ_m = a random variable with unit median

Random variable (RV) ϵ_m represents the variability resulting from site-to-site differences in the local meteorology. The distribution of ϵ_m was constructed so the RV has a median value of 1.0 and the shape of the distribution was selected to approximate the results from the Sandia siting study. The distribution was selected to be lognormal with a 90 percentile probability interval (95th/5th) equal to a factor of 2 (Error Factor = 1.41). The distribution on EI_i was then determined by Monte Carlo simulation, assuming EI_r and ϵ_m are independent.

The distribution for the exposure index in Equation 4 is shown in Figure 3. The statistics of the distribution are shown in Table 2. The distribution has a mean value of 0.063, a median value of 0.057 and a 95th percentile value of 0.126. The ratio of the 95th percentile value to the mean value is 2.0.

Procedure for Estimating Subsidiary LERF Criteria

An approach to deriving subsidiary LERF criteria from the safety goal QHO for individual early fatality risk is discussed below. We start with Equation 1 and recognize that this is generally not an exact equality. More accurately stated:

$$\begin{aligned} IEFF &\propto LERF \times EI \\ \text{or} \\ IEFF &= \frac{1}{C} \times LERF \times EI \end{aligned} \quad \text{Equation 5}$$

where: C = proportionality constant

To determine the value of C we must first of all have a precise physical definition of LERF in terms of the characteristics described earlier in the section entitled "Parameters Important to Early Fatality Risk." With this definition in hand, the value of the constant C can be estimated using the (mean values for) LERF and IEFF from existing Level 3 probabilistic risk assessments (PRAs) such as NUREG-1150 and site-specific exposure index values from this study (or calculated using the same population distribution and wind direction frequency input information used in the PRAs for the consequence analyses).

Rearranging Equation 5 and substituting the value of the safety goal QHO for individual early fatality risk (QHO_{IEFF}) for the IEFF yields the following relationship for deriving the subsidiary LERF criteria ($LERF_c$):

$$LERF_c = C \times \frac{QHO_{IEFF}}{EI} \quad \text{Equation 6}$$

To account for variability in the site parameters important to individual early fatality risk, the constant C can be adjusted to bound the results for the population of sites. For example, on the basis of the results from this study indicating that the 95th percentile value of the EI was two times the mean value, the value of C could be reduced by a factor of 2 to bound this variability (at the 95th percentile level). This would result in a reduction in the LERF criteria by a factor of 2.

Conclusions and Recommendations

The range of variability due to site-to-site population distribution, wind direction frequency distribution, exclusion zone size, and meteorology record has been evaluated. The ratio of the 95th percentile value to the mean value of the exposure index (EI) is approximately 2.

Since this variability is much less than the magnitude of the other uncertainties associated with the (Level 1 and Level 2 PRA) calculation of the LERF, it is judged that there is insufficient site-to-site variability in individual early fatality risk to warrant development of site-specific subsidiary LERF criteria. Hence, a single lower-level (LERF) criterion can be determined on a generic basis.

The information provided on the variability of the individual early fatality risk resulting from site-related parameters can be used to develop an appropriate factor to bound the variability.

Derivation of a subsidiary LERF criterion, however, cannot proceed until a physical definition of the LERF is developed in terms of the parameters, important to early fatality risk, which have been identified above.

References

- (ACRS-1996) "Plant-Specific Application of Safety Goals," Report to Shirley A. Jackson, Chairman, U.S. Nuclear Regulatory Commission from T. S. Kress, Chairman, Advisory Committee on Reactor Safeguards, November 18, 1996.
- (ACRS-1997) "Risk-Based Regulatory Acceptance Criteria for Plant-Specific Application of Safety Goals," Report to Shirley A. Jackson, Chairman,

U.S. Nuclear Regulatory Commission from R. L. Seale, Chairman, Advisory Committee on Reactor Safeguards, April 11, 1997.

- (NRC-1982) Aldrich, D.C., et al, "Technical Guidance for Siting Criteria Development," NUREG/CR-2239, December, 1982.
- (Sherry-1997) "Methodology for Estimating Offsite Early Fatality Risk in the Absence of a Level 3 PRA," Attachment 2 to (ACRS,1997)
- (SRM-1997a) "Staff Requirement - COMSECY-96-01 - Risk-Informed, Performance-Based Regulation (DSI-12)," Memorandum to L. Joseph Callan, Executive Director for Operations, and Karen D. Cyr, General Counsel, from Annette L. Vietti-Cook, Acting Secretary, U.S. Nuclear Regulatory Commission, dated April 15, 1997.
- (SRM-1997b) "Staff Requirement - Meeting with the Advisory Committee on Reactor Safeguards," Memorandum to John T. Larkins, Executive Director ACRS, from John Hoyle, Secretary, U.S. Nuclear Regulatory Commission, dated May 27, 1997.

Table 1 - Site Specific Exposure Indices (Step 1 and Step 2)

Site	Step 1	Step 2
	EI	EI_r
Arkansas 1,2	0.077	0.063
Beaver Valley 1,2	0.066	0.082
Big Rock Point	0.051	0.051
Browns Ferry 1,2,3	0.081	0.058
Braidwood 1,2	0.077	0.114
Brunswick 1,2	0.060	0.055
Byron 1,2	0.059	0.086
Callaway	0.098	0.078
Calvert Cliffs 1,2	0.040	0.030
Catawba 1,2	0.052	0.055
Clinton	0.086	0.075
Comanche Peak 1,2	0.043	0.027
D.C. Cook 1,2	0.061	0.075
Cooper	0.033	0.036
Crystal River 3	0.045	0.029
Davis-Besse	0.045	0.055
Diablo Canyon 1,2	0.063	0.063
Dresden 2,3	0.079	0.091
Duane Arnold	0.056	0.085
Farley 1,2	0.061	0.043
Fermi 2	0.035	0.032
Fitzpatrick	0.053	0.046
Fort Calhoun	0.079	0.130

Ginna	0.045	0.066
Grand Gulf	0.077	0.082
Hatch 1,2	0.062	0.043
Indian Point 2,3	0.078	0.134
Kewaunee	0.040	0.029
LaSalle 1,2	0.045	0.062
Limerick 1,2	0.071	0.076
Maine Yankee	0.056	0.069
McQuire 1,2	0.052	0.055
Millstone 1,2,3	0.081	0.113
Monticello	0.066	0.093
Nine Mile Point 1,2	0.055	0.031
North Anna 1,2	0.076	0.049
Oconee 1,2,3	0.072	0.040
Oyster Creek	0.082	0.129
Palisades	0.058	0.066
Palo Verde 1,2,3	0.052	0.048
Peach Bottom 2,3	0.047	0.047
Perry	0.055	0.050
Pilgrim	0.046	0.069
Point Beach 1,2	0.052	0.038
Praire Island 1,2	0.094	0.105
Quad Cities 1,2	0.068	0.109
Riverbend	0.065	0.060
H. B. Robinson	0.062	0.095
St. Lucie 1,2	0.042	0.024

Salem 1,2/Hope Creek	0.063	0.047
San Onofre 2,3	0.070	0.071
Seabrook	0.052	0.048
Sequoyah 1,2	0.022	0.028
Shearon Harris	0.062	0.025
South Texas 1,2	0.115	0.071
Summer	0.064	0.035
Surry 1,2	0.056	0.073
Susquehanna 1,2	0.062	0.081
Three Mile Island	0.066	0.081
Turkey Point 3,4	0.077	0.053
Vermont Yankee	0.040	0.075
Vogtle 1,2	0.063	0.049
Waterford 3	0.071	0.065
Watts Bar	0.042	0.030
WNP-2	0.063	0.028
Wolf Creek	0.055	0.040
Zion 1,2	0.039	0.036

Table 2 - Summary of Exposure Index Distribution Statistics

	Step 1	Step 2	Step 3
Statistic	EI	EI _r	EI _t
mean	.0609	.0623	.0633
median	.0611	.0582	.0570
5th percentile	.0394	.0282	.0249
95th percentile	.0849	.1136	.1260
97.5 percentile	-	-	.1423
standard deviation	.0166	.0273	.0314

cc: J. Larkins
R. Savio
S. Duraiswamy
A. Cronenberg
ACRS Senior Engineers

Figure 1 - Exposure Indices - Step 1

Figure 2 - Exposure Indices - Step 2

Figure 3 - Exposure Index - Step 3

Cell D7

Frequency Chart 10,000 Trials Shown

Attachment 1 - Site Coordinates

Site	Location	State	Latitude	Longitude
ARKANSAS NUCLEAR	RUSSELLVILLE	AR	351336	931351
BEAVER VALLEY	SHIPPINGPORT	PA	403719	802602
BIG ROCK POINT	CHARLEVOIX	MI	452133	851141
BRAIDWOOD	BRACEVILLE	IL	411437	881344
BROWNS FERRY	DECATUR	AL	344215	870707
BRUNSWICK	SOUTHPORT	NC	335730	780038
BYRON	BYRON	IL	420430	891655
CALLAWAY	FULTON	MO	384530	914654
CALVERT CLIFFS	LUSBY	MD	382605	762631
CATAWBA	YORK	SC	350305	810410
CLINTON	CLINTON	IL	401019	885003
COMANCHE PEAK	GLEN ROSE	TX	321752	974706
COOK	BRIDGMAN	MI	415834	863359
COOPER STATION	BROWNSVILLE	NE	402143	953828
CRYSTAL RIVER	CRYSTAL RIVER	FL	285726	824156
DAVIS-BESSE	OAK HARBOR	OH	413550	830511
DIABLO CANYON	AVILA BEACH	CA	351242	1205116
DRESDEN	MORRIS	IL	412323	881616
DUANE ARNOLD	PALO	IA	420602	914638
FARLEY	ASHFORD	AL	311322	850645
FERMI	NEWPORT	MI	415748	831531
FITZPATRICK	LYCOMING	NY	433126	762354
FORT CALHOUN	FORT CALHOUN	NE	413115	960436
GINNA	ONTARIO	NY	431690	771832
GRAND GULF	PORT GIBSON	MS	320027	910253
HADDAM NECK	HARTFORD	CT	412855	722957
HARRIS	RALEIGH	NC	353800	785722
HATCH	BAXLEY	GA	315603	822040
HOPE CREEK	HANCOCKS BRIDGE	NJ	392804	753217
HUMBOLDT BAY	SAN FRANCISCO	CA	404431	12412 0

INDIAN POINT	BUCHANAN	NY	411617	735709
KEWAUNEE	KEWAUNEE	WI	442035	873210
LASALLE	MARSEILLES	IL	411438	884015
LIMERICK	PHILADELPHIA	PA	401312	753524
MAINE YANKEE	WISCASSET	ME	435702	694146
MCGUIRE	CORNELIUS	NC	352556	805654
MILLSTONE	WATERFORD	CT	411831	721005
MONTICELLO	MONTICELLO	MN	452000	935054
NINE MILE POINT	LYCOMING	NY	433120	762436
NORTH ANNA	RICHMOND	VA	380339	774726
OCONEE	SENECA	SC	344730	825355
OYSTER CREEK	FORKED RIVER	NJ	394851	741223
PALISADES	COVERT	MI	421920	861855
PALO VERDE	WINTERSBURG	AZ	332323	1125143
PEACH BOTTOM	PHILADELPHIA	PA	394532	761609
PERRY	PERRY	OH	414804	810836
PILGRIM	PLYMOUTH	MA	415640	703446
POINT BEACH	TWO RIVERS	WI	441651	873210
PRAIRIE ISLAND STATION	WELCH	MN	443710	923759
QUAD CITIES	CORDOVA	IL	414334	901836
RANCHO SECO	HERALD	CA	382046	1210708
RIVER BEND STATION	ST FRANCISVILLE	LA	304526	911954
ROBINSON	HARTSVILLE	SC	342419	800931
SALEM	HANCOCKS BRIDGE	NJ	392746	753209
SAN ONOFRE	SAN CLEMENTE	CA	332213	1173325
SEABROOK	MANCHESTER	NH	425353	705105
SEQUOYAH	SODDY-DAISY	TN	351324	850516
SOUTH TEXAS PROJECT	WADSWORTH	TX	284742	960253
ST LUCIE	JENSEN BEACH	FL	272055	801447
SUMMER	JENKINSVILLE,	SC	341745	811913
SURRY	SURRY	VA	370956	764154
SUSQUEHANNA	ALLENTOWN	PA	410530	760855

THREE MILE ISLAND	MIDDLETOWN	PA	400911	764330
TURKEY POINT	MIAMI	FL	252606	801953
VERMONT YANKEE	VERNON	VT	424649	723057
VOGTLE	WAYNESBORO	GA	330831	814553
WATERFORD	KILLONA	LA	295942	902816
WATTS BAR	SPRING CITY	TN	353610	844725
WNP	RICHLAND	WA	462817	1191959
WOLF CREEK STATION	BURLINGTON	KS	381420	954120
ZION	ZION	IL	422644	874808

Attachment 2

Site Wind Rose Data - from NUREG/CR-2239

Table A.4-1 Site Wind Rose Data
Probability of Wind Blowing Towards Sector

Station	N S	ENE SSW	NE SW	NNE WSW	E W	ESE WSW	SE SW	SEE SSW
Allens Creek		<u>10 m</u>			<u>8/1/1972 - 7/31/1973</u>			
	.121	.073	.043	.024	.022	.021	.027	.069
	.107	.075	.062	.030	.046	.053	.101	.104
Arkansas 1, 2		<u>190 ft</u>			<u>6/69 - 5/70</u>			
	.103	.074	.052	.074	.126	.087	.053	.021
	.025	.015	.037	.036	.098	.077	.037	.042
Bailly		<u>230 ft</u>			<u>12/4/51 - 12/3/57</u>			
	.064	.105	.095	.086	.069	.056	.040	.038
	.064	.068	.069	.063	.038	.028	.033	.066
Beaver Valley 1, 2		<u>150 ft</u>			<u>9/15/49 - 9/3/70</u>			
	.087	.070	.051	.041	.083	.137	.123	.030
	.035	.023	.021	.023	.040	.059	.067	.064
Bellefonte 1		<u>84 ft</u>			<u>1971</u>			
	.064	.075	.092	.082	.071	.067	.060	.076
	.069	.066	.031	.040	.037	.053	.064	.053
Big Rock Point		<u>250 ft</u>			<u>2/61 - 2/63</u>			
	.112	.075	.071	.081	.099	.058	.057	.065
	.056	.039	.034	.046	.088	.037	.037	.045
Black Fox		<u>33 ft</u>			<u>12/73 - 11/74</u>			
	.180	.055	.026	.026	.022	.030	.051	.059
	.067	.064	.056	.045	.034	.046	.079	.160
Braidwood 1		<u>30 ft</u>			<u>12/1/73 - 10/31/74</u>			
	.105	.113	.077	.065	.061	.070	.065	.045
	.052	.048	.048	.045	.043	.044	.056	.063
Browns Ferry 1, 2, 3		<u>200 ft</u>			<u>2/11/67 - 12/31/68</u>			
	.072	.066	.058	.058	.052	.067	.055	.054
	.052	.067	.056	.038	.032	.072	.101	.099
Brunswick 1, 2		<u>350 ft</u>			<u>9/25/70 - 1/3/73</u>			
	.055	.077	.145	.088	.053	.037	.036	.041
	.059	.065	.084	.078	.053	.044	.047	.038
Byron 1		<u>30 ft</u>			<u>6/11/73 - 5/31/74</u>			
	.097	.089	.081	.065	.075	.063	.076	.057
	.053	.037	.048	.058	.049	.044	.039	.069
Calleyway		<u>10 m</u>			<u>5/4/73 - 5/4/74</u>			
	.126	.096	.074	.043	.058	.070	.058	.050
	.051	.040	.026	.028	.036	.046	.083	.116
Calvert Cliff 1, 2		<u>33 ft</u>						
	.116	.089	.070	.045	.064	.061	.103	.078
	.084	.058	.038	.024	.035	.028	.028	.082
Catawba 1		<u>30 ft</u>			<u>6/30/71 - 6/30/72</u>			
	.023	.056	.207	.087	.043	.024	.026	.026
	.075	.079	.179	.060	.033	.025	.040	.017
Cherokee		<u>30 ft</u>			<u>9/11/73 - 9/11/74</u>			
	.036	.048	.124	.104	.094	.081	.114	.059
	.064	.059	.075	.055	.029	.022	.036	.019
Clinton		<u>10 m</u>			<u>5/72 - 4/73</u>			
	.104	.093	.086	.054	.042	.041	.042	.052
	.070	.068	.071	.054	.034	.038	.049	.067
Comanche Peak		<u>10 m</u>			<u>5/12/72 - 5/14/76</u>			
	.151	.084	.041	.025	.024	.029	.067	.076
	.060	.040	.029	.025	.032	.069	.105	.149

Table A.4-1 Site Wind Rose Data (cont)
Probability of Wind Blowing Towards Sector

Station	N S	NNE SEW	NE SW	NNE WSW	E W	ESE WNW	SE NW	ESE WNW
Cook DC 1, 2		<u>200 ft</u>				<u>1967</u>		
	.091 .078	.105 .042	.055 .042	.045 .050	.056 .040	.069 .063	.057 .072	.062 .073
Cooper		<u>318 ft</u>				<u>3/70 - 2/71</u>		
	.116 .094	.117 .061	.079 .025	.037 .031	.030 .027	.041 .034	.060 .058	.100 .090
Crystal River		<u>33 ft</u>				<u>1/1/75 - 12/31/75</u>		
	.043 .062	.048 .047	.051 .098	.048 .121	.082 .111	.057 .064	.043 .061	.030 .034
Davis-DE 1		<u>35 ft</u>				<u>8/4/74 - 8/3/75</u>		
	.064 .030	.116 .039	.130 .058	.102 .057	.081 .077	.039 .041	.053 .038	.037 .039
Diablo Canyon 1, 2		<u>250 ft</u>				<u>10/69 - 9/70</u>		
	.031 .059	.012 .029	.014 .055	.015 .017	.026 .014	.045 .015	.363 .103	.128 .075
Dresden 2, 3		<u>300 ft</u>						
	.088 .049	.090 .031	.096 .039	.067 .033	.101 .036	.085 .033	.080 .060	.056 .055
Duane Arnold		<u>165 ft</u>				<u>1971</u>		
	.129 .075	.073 .040	.053 .032	.036 .034	.051 .039	.062 .060	.083 .061	.095 .076
Farley 1, 2		<u>33 ft</u>						
	.073 .097	.070 .083	.064 .086	.044 .062	.044 .044	.045 .035	.067 .040	.090 .056
Fermi 2						<u>9/1/73 - 8/31/74</u>		
	.041 .026	.088 .025	.089 .059	.102 .063	.083 .069	.086 .050	.063 .058	.047 .058
Fitzpatrick		<u>200 ft</u>				<u>1963 - 1964</u>		
	.087 .040	.059 .047	.102 .033	.132 .014	.115 .018	.056 .037	.053 .101	.035 .068
Forked River		<u>400 ft</u>				<u>2/66 - 2/67</u>		
	.075 .044	.096 .037	.087 .052	.068 .055	.087 .039	.093 .040	.075 .047	.063 .040
Fort Calhoun		<u>40 ft</u>				<u>10/68 - 9/70</u>		
	.093 .071	.059 .018	.034 .017	.021 .022	.042 .028	.079 .064	.113 .115	.098 .126
Fort St. Vrain		<u>205 ft</u>				<u>1967 - 1968</u>		
	.063 .164	.069 .085	.076 .076	.057 .064	.040 .058	.029 .043	.035 .051	.039 .049
Ginna R.E.		<u>50 ft</u>				<u>1966 - 1967</u>		
	.090 .030	.081 .032	.102 .031	.097 .038	.112 .045	.101 .036	.079 .030	.044 .052
Grand Gulf 1						<u>1951 - 1960</u>		
	.101 .065	.074 .060	.062 .061	.043 .040	.036 .040	.043 .044	.070 .080	.064 .117
Hadden Neck		<u>129 ft</u>				<u>1963</u>		
	.048 .013	.046 .006	.043 .009	.038 .013	.070 .035	.160 .092	.265 .055	.052 .055
Hertsville		<u>33 ft</u>				<u>2/1/73 - 1/31/74</u>		
	.045 .045	.058 .113	.048 .175	.056 .063	.051 .050	.034 .074	.044 .069	.025 .051

Table A.4-1 Site Wind Rose Data (cont)
Probability of Wind Blowing Towards Sector

Station	N S	NNE SSW	NE SW	ENE WSW	E W	ESE WNW	SE WN	SSE WNW
Batch, E.I. 1, 2		<u>150 ft</u>			<u>6/1/70 - 8/31/74</u>			
	.055 .040	.069 .038	.082 .051	.073 .067	.075 .081	.077 .068	.072 .057	.049 .044
Indian Point 2, 3		<u>100 ft</u>			<u>1/1/71 - 12/31/71</u>			
	.076 .124	.055 .135	.038 .066	.039 .027	.053 .019	.079 .019	.077 .041	.070 .063
Kewaunee		<u>180 ft</u>			<u>8/31/68 - 3/25/70</u>			
	.082 .066	.090 .055	.064 .042	.075 .030	.094 .022	.117 .023	.082 .028	.080 .050
LaSalle 1, 2		<u>300 ft</u>						
	.088 .049	.090 .031	.096 .039	.067 .033	.101 .036	.085 .033	.080 .060	.056 .055
Le Crosse		<u>350 ft</u>			<u>1968 - 1970</u>			
	.194 .125	.139 .101	.084 .048	.018 .011	.051 .022	.026 .010	.076 .026	.062 .033
Limerick 1		<u>270 ft</u>			<u>1/72 - 12/74</u>			
	.071 .054	.068 .039	.052 .035	.051 .046	.090 .0670	.150 .040	.109 .037	.059 .040
Marble Hill		<u>33 ft</u>			<u>1/74 - 12/74</u>			
	.058 .045	.141 .044	.124 .063	.074 .060	.062 .047	.060 .030	.044 .030	.037 .041
Mc Yankee		<u>149 ft</u>			<u>7/67 - 6/68</u>			
	.118 .075	.124 .068	.082 .064	.041 .030	.041 .024	.055 .027	.088 .031	.089 .044
McGuire 1, 2		<u>130 ft</u>			<u>10/17/70 - 10/16/71</u>			
	.070 .057	.090 .068	.122 .113	.062 .078	.054 .056	.042 .037	.042 .038	.040 .030
Midland 2					<u>1962 - 1966</u>			
	.060 .045	.082 .046	.123 .061	.106 .043	.124 .045	.066 .024	.064 .028	.051 .032
Millstone 1, 2		<u>152 ft</u>			<u>8/65 - 9/67</u>			
	.038 .066	.060 .060	.076 .036	.170 .035	.078 .058	.070 .035	.078 .025	.073 .041
Monticello		<u>140 ft</u>			<u>2/9/67 - 2/10/68</u>			
	.089 .036	.091 .041	.063 .029	.055 .051	.030 .031	.089 .055	.104 .052	.118 .065
Nine M. Pt. 1, 2		<u>204 ft</u>			<u>1963 - 1964</u>			
	.082 .041	.060 .048	.104 .034	.131 .013	.118 .018	.059 .037	.054 .097	.037 .069
North Anna 1, 2, 3		<u>150 ft</u>			<u>9/16/71 - 9/15/72</u>			
	.141 .100	.095 .048	.058 .044	.047 .035	.055 .041	.047 .035	.074 .042	.084 .054
Oconee 1, 2, 3					<u>6/19/68 - 6/19/69</u>			
	.021 .174	.036 .084	.075 .100	.051 .058	.062 .060	.043 .038	.061 .036	.081 .019
Oyster Creek		<u>400 ft</u>			<u>2/66 - 2/67</u>			
	.075 .044	.096 .037	.087 .052	.068 .055	.087 .039	.093 .040	.075 .047	.063 .040
Palisade		<u>35 ft</u>			<u>9/67 - 8/68</u>			
	.204 .080	.113 .033	.027 .013	.030 .012	.058 .052	.046 .038	.072 .049	.081 .093

Table A.4-1 Site Wind Rose Data (cont)
Probability of Wind Blowing Towards Sector

Station	N S	NNE SSW	NE SE	ENE WSW	E W	ESE WNW	SE NW	SSE NNW
Palo Verde 1		<u>200 ft</u>			<u>8/13/73 - 8/13/74</u>			
	.055	.073	.144	.082	.068	.047	.052	.035
	.048	.059	.068	.048	.073	.059	.056	.041
Peach Bottom 2, 3		<u>320 ft</u>			<u>8/67 - 7/69</u>			
	.085	.064	.046	.052	.069	.095	.115	.109
	.060	.043	.031	.032	.034	.046	.054	.064
Pebble Springs		<u>30 ft</u>			<u>1/74 - 12/74</u>			
	.017	.039	.075	.201	.313	.094	.021	.009
	.012	.019	.050	.055	.035	.028	.020	.014
Perkins		<u>30 ft</u>			<u>10/12/73 - 10/11/74</u>			
	.036	.067	.125	.066	.058	.047	.064	.053
	.068	.066	.104	.067	.063	.037	.044	.034
Perry 1		<u>200 ft</u>			<u>5/1/72 - 4/30/73</u>			
	.105	.095	.082	.084	.081	.054	.057	.042
	.045	.030	.057	.045	.048	.037	.054	.073
Phipps Bend		<u>33 ft</u>			<u>2/1/74 - 1/31/75</u>			
	.037	.054	.107	.106	.053	.071	.053	.120
	.054	.110	.112	.045	.020	.018	.021	.019
Pilgrim 1		<u>72 ft</u>						
	.051	.185	.118	.085	.094	.060	.053	.046
	.051	.038	.042	.035	.048	.031	.033	.030
Point Beach 1, 2		<u>150 ft</u>			<u>4/67 - 4/68</u>			
	.088	.122	.087	.048	.081	.097	.075	.056
	.096	.070	.055	.022	.020	.018	.031	.036
Prairie 1, 2		<u>140 ft</u>			<u>6/1/71 - 5/31/72</u>			
	.065	.031	.025	.031	.073	.102	.125	.065
	.046	.023	.019	.019	.055	.108	.134	.080
Quad Cities 1, 2		<u>400 ft</u>			<u>4/68 - 9/69</u>			
	.072	.128	.090	.049	.045	.069	.083	.067
	.068	.051	.042	.028	.037	.033	.075	.063
Rancho Seco		<u>50 ft</u>			<u>1967 - 1969</u>			
	.066	.073	.069	.107	.114	.078	.100	.074
	.049	.034	.029	.021	.029	.039	.057	.062
Riverbend 1		<u>135 ft</u>			<u>10/1/72 - 9/30/73</u>			
	.057	.058	.048	.048	.054	.048	.061	.066
	.069	.066	.066	.060	.076	.082	.072	.067
M. B. Robinson 2		<u>120 ft</u>			<u>4/14/67 - 4/19/68</u>			
	.045	.074	.072	.081	.071	.037	.036	.043
	.141	.114	.095	.050	.040	.035	.038	.029
Saint Lucie 1		<u>50 ft</u>			<u>11/1/72 - 12/31/72</u>			
	.062	.056	.063	.046	.030	.041	.053	.029
	.045	.038	.070	.088	.121	.093	.098	.067
Salem 1, 2		<u>300 ft</u>			<u>6/69 - 5/71</u>			
	.067	.062	.060	.056	.073	.095	.132	.094
	.062	.046	.049	.037	.028	.023	.042	.074
San Onofre		<u>10 m</u>			<u>1/25/73 - 1/24/76</u>			
	.066	.061	.054	.065	.088	.109	.060	.031
	.034	.111	.134	.028	.016	.022	.049	.070
Seabrook 1		<u>30 ft</u>			<u>11/71 - 10/72</u>			
	.030	.040	.069	.089	.110	.167	.145	.049
	.039	.024	.033	.046	.038	.041	.043	.037

Table A.4-1 Site Wind Rose Data (cont)
Probability of Wind Blowing Towards Sector

Station	N E	NNE ENE	NE SE	NNE ENE	E W	ESE WNW	SE SW	SSE SSW
Sequoyah 1, 2		<u>33 ft</u>			<u>4/21/71 - 3/31/72</u>			
	.066 .058	.151 .149	.161 .116	.48 .026	.024 .011	.024 .008	.035 .013	.070 .019
Shearon Harris		<u>10 m</u>			<u>1/76 - 12/76</u>			
	.079 .083	.107 .067	.098 .063	.079 .047	.053 .033	.034 .031	.057 .035	.062 .053
Skagit		<u>10 m</u>						
	.014 .037	.011 .021	.021 .041	.037 .028	.128 .109	.109 .058	.085 .039	.062 .020
Shoreham		<u>150 ft</u>			<u>10/1/73 - 9/30/74</u>			
	.060 .050	.129 .045	.095 .049	.050 .043	.079 .032	.103 .028	.094 .036	.066 .041
South Texas		<u>33 ft</u>			<u>7/20/73 - 7/20/74</u>			
	.148 .075	.046 .078	.029 .080	.010 .047	.015 .053	.014 .059	.020 .137	.037 .153
Virgin C. Summer		<u>202 ft</u>			<u>1975</u>			
	.068 .029	.090 .042	.118 .080	.087 .070	.064 .059	.046 .041	.055 .052	.043 .056
Wurry St 1, 2		<u>150 ft</u>			<u>11/67 - 12/69</u>			
	.064 .072	.082 .051	.082 .046	.062 .045	.059 .057	.061 .052	.087 .055	.081 .043
Susquehanna 1					<u>1956 - 1960</u>			
	.037 .046	.070 .039	.125 .049	.126 .054	.044 .040	.059 .062	.100 .031	.090 .029
Three Mile Island		<u>100 ft</u>			<u>4/71 - 3/72</u>			
	.054 .040	.045 .027	.054 .036	.059 .057	.091 .085	.092 .082	.091 .062	.070 .057
Trojan		<u>30 ft</u>			<u>9/1/71 - 8/31/72</u>			
	.203 .172	.070 .054	.026 .016	.013 .006	.022 .007	.037 .009	.070 .046	.132 .120
Turkey Point 1, 2		<u>30 ft</u>			<u>1969</u>			
	.038 .035	.041 .028	.047 .048	.027 .100	.027 .136	.047 .135	.051 .100	.077 .062
Vermont Yankee 1		<u>140 ft</u>			<u>9/67 - 7/68</u>			
	.072 .070	.027 .025	.018 .017	.023 .019	.069 .024	.086 .066	.117 .085	.196 .086
Vogtle		<u>30 ft</u>			<u>12/73 - 12/74</u>			
	.064 .043	.062 .043	.074 .072	.079 .065	.084 .069	.075 .060	.056 .063	.031 .060
Waterford 3		<u>30 ft</u>			<u>5/72 - 4/73</u>			
	.042 .046	.053 .092	.045 .088	.047 .059	.049 .029	.056 .100	.064 .083	.072 .077
Watts Bar 1, 2		<u>200 ft</u>			<u>7/1/73 - 6/30/75</u>			
	.033 .053	.109 .106	.183 .132	.089 .059	.040 .041	.031 .019	.035 .014	.037 .019
WPPS 1, 4		<u>33 ft</u>			<u>4/74 - 3/75</u>			
	.180 .164	.082 .045	.063 .036	.052 .031	.061 .022	.099 .026	.107 .040	.085 .075
WPPS 2		<u>33 ft</u>			<u>4/74 - 3/75</u>			
	.180 .164	.082 .045	.063 .036	.052 .031	.061 .022	.099 .026	.107 .040	.085 .075

Table A.4-1 Site Wind Rose Data (cont)
Probability of Wind Blowing Towards Sector

Station	N S	NNE SSW	NE SW	ENE WSW	E W	ESE WNW	SE NNW	SSE NNN
WPPS 3, 5		<u>60 m</u>			<u>5/73 - 4/74</u>			
	.071	.098	.124	.170	.125	.031	.015	.010
	.014	.019	.062	.074	.047	.052	.050	.027
Wolf Creek		<u>10 m</u>			<u>6/1/73 - 5/31/75</u>			
	.080	.100	.040	.024	.030	.041	.064	.069
	.164	.058	.039	.035	.039	.046	.061	.111
Yankee Rowe		<u>30 ft</u>			<u>10/71 - 9/72</u>			
	.101	.080	.052	.037	.039	.041	.072	.086
	.086	.064	.065	.063	.047	.036	.052	.081
Yellow Creek		<u>33 ft</u>			<u>7/1/74 - 6/30/75</u>			
	.142	.097	.049	.039	.040	.050	.057	.087
	.037	.070	.049	.019	.021	.046	.060	.130
Zimmer 1		<u>30 ft</u>			<u>3/1/72 - 2/28/74</u>			
	.108	.066	.068	.056	.051	.059	.047	.062
	.062	.031	.027	.023	.030	.054	.127	.129
Zion		<u>35 ft</u>			<u>1970</u>			
	.071	.078	.079	.113	.069	.076	.046	.071
	.046	.059	.037	.039	.035	.035	.060	.096

Attachment 3

Site Minimum Exclusion Zone Distances - from NUREG/CR-2239

TABLE D.2-1

EXCLUSION DISTANCES (MILES) FOR 91 REACTOR SITES

SITE	EX. DIST.
*****	*****
1 ALLENS CREEK	0.82
2 ARKANSAS 1 + 2	0.65
3 BAILLY S	0.12
4 BEAVER VALLEY 1 + 2	0.38
5 BELLEFONTE 1	0.57
6 BIG ROCK POINT	0.51
7 BLACK FOX	0.53
8 BRAIDWOOD 1	0.28
9 BROWNS FERRY 1, 2, + 3	0.76
10 BRUNSWICK 1 + 2	0.57
11 BYRON 1	0.29
12 CALLAWAY	0.68
13 CALVERT CLIFF 1 + 2	0.71
14 CATAWBA 1	0.47
15 CHEROKEE	0.37
16 CLINTON	0.61
17 COMMANCHE PEAK	0.87
18 COOK DC 1 + 2	0.38
19 COOPER S	0.46
20 CRYSTAL RIVER	0.83
21 DAVIS-BE 1	0.39
22 DIABLO CANYON 1 + 2	0.50
23 DRESDEN 2 + 3	0.42
24 DUANE ARNOLD	0.27
25 FARLEY 1 + 2	0.78
26 FERMI 2	0.57
27 FITZPATRICK	0.61
28 FORKED RIVER 1	0.38
29 FORT CALHOUN	0.23
30 FORT ST VRAIN	0.37
31 R. E. GINNA	0.28
32 GRAND GULF 1	0.47
33 HADDEM NECK	0.33
34 HARTSVILLE	0.76
35 HATCH, E.I. 1 + 2	0.78
36 INDIAN PT 2 + 3	0.21
37 KEWAUNEE	0.75
38 LASALLE 1 + 2	0.32
39 LA CROSSE	0.21
40 LIMERICK 1	0.47
41 MARBLE HILL	0.42
42 ME YANKEE	0.38
43 MCGUIRE 1 + 2	0.47
44 MIDLAND 2	0.31
45 MILLSTONE 1 + 2	0.31

TABLE D.2-1 (cont'd)

SITE	EX. DIST.
*****	*****
46 MONTICELLO	0.30
47 NINE M. PT. 1 + 2	0.97
48 NORTH ANNA 1, 2, + 3	0.84
49 OCONEE 1, 2 + 3	1.00
50 OYSTER CREEK	0.25
51 PALISADE	0.42
52 PALO VERDE 1	0.56
53 PEACH BOTTOM 2 + 3	0.51
54 PEBBLE SPRINGS	0.49
55 PERKINS	0.37
56 PERRY 1	0.57
57 PHIPPS BEND	0.47
58 PILGRIM 1	0.27
59 POINT BEACH 1 + 2	0.75
60 PRAIRIE 1 + 2	0.44
61 QUAD CITIES 1 + 2	0.24
62 RANCHO SECO	0.40
63 RIVERBEND 1	0.57
64 H. B. ROBINSON 2	0.26
65 SAINT LUCIE 1	0.97
66 SALEM 1 + 2	0.72
67 SAN ONOFRE	0.50
68 SEABROOK 1	0.57
69 SEQUOYAH 1 + 2	0.36
70 SHEARON HARRIS	1.33
71 SHOREHAM	0.19
72 SKAGIT	0.38
73 SOUTH TEXAS	0.89
74 VIRGIL C. SUMMER	1.01
75 SURRY ST 1 + 2	0.35
76 SUSQUEHANNA 1	0.35
77 THREE MILE ISLAND	0.38
78 TROJAN	0.41
79 TURKEY POINT 1 + 2	0.79
80 VERMONT YANKEE 1	0.17
81 VOGTLE	0.68
82 WATERFORD 3	0.57
83 WATTS BAR 1 + 2	0.75
84 WPPSS1+4	1.21
85 WPPSS 3 + 5	0.81
86 WPPSS 2	1.21
87 WOLF CREEK	0.75
88 YANKEE ROWE	0.59
89 YELLOW CREEK	0.43
90 ZIMMER 1	0.24
91 ZION	0.57

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

June 11, 1997

MEMORANDUM TO: ACRS Members

FROM: Rick Sherry, Senior ACRS Fellow *Rick Sherry*

SUBJECT: CONSIDERATIONS OF SOCIETAL RISK IN PLANT-SPECIFIC, SITE-SPECIFIC APPLICATION OF SAFETY GOALS AND DEFINITION OF SUBSIDIARY CRITERIA

Background

In its November 18, 1996 report (ACRS-1996), the ACRS stated that "the safety goals and subsidiary objectives can and should be used to derive guidelines for plant-specific applications." In its April 11, 1997 report (ACRS-1997), the ACRS stated further that "Quantification of the LERF [large, early release frequency] at each site is needed to ensure the appropriateness of the choice of the LERF acceptance criterion proposed in draft Regulatory Guide DG-1061 and draft Standard Review Plan sections that support risk-informed, performance-based regulation."

In a Staff Requirements Memorandum dated April 15, 1997 (SRM-1997), the Commission stated (referring to *Direction Setting Issue 12*), "The staff should develop objective standard(s) for the application of risk-informed, performance-based and risk-informed less prescriptive approaches to regulations on an expedited basis. Such standard(s) could be in the form of individual plant safety goals and subsidiary objective performance criteria as discussed in the issue paper."

Issue

An important issue that arises when considering application of the safety goals on a plant-specific, site-specific basis is whether individual risk alone is adequate for characterizing risk or should other risk metrics, such as societal risk, be considered in assessing risk-informed regulatory applications?

ATTACHMENT 2

Discussion

The two quantitative health objectives (QHOs) delineated in the NRC Safety Goal policy statement (NRC-1986) are both stated in terms of individual risk. Because of this formulation of the QHOs, and as pointed out many years ago by the ACRS (ACRS-1983), "Larger societal risks are permitted for the nuclear power plant which has the larger surrounding population...." The ACRS has proposed in the past that a societal risk goal be applied to early fatalities (ACRS-1980). Recently, other countries have developed nuclear power plant quantitative safety criteria that explicitly include societal as well as individual risk (Versteeg-1992).

Contained in the NRC Safety Goal Policy Statement are separate views of Commissioner Bernthal, which include the following:

The absence of such explicit population density considerations in the Commission's 0.1 percent goals for offsite consequences deserves careful thought. Is it reasonable that Zion and Palo Verde, for example, be assigned the same 'standard person' risk, even though they pose considerably different risks for the U.S. population as a whole? As they stand these 0.1 percent goals do not explicitly include population density considerations.

Although it may have been acceptable to neglect explicit considerations of societal risk in the safety goals when the goals were to be used in a generic fashion to assess the risk of the population of plants as a whole, it is less clear that this is acceptable when the goals are used to assess the acceptability of proposed changes on a plant-specific basis.

The Sandia siting study (NRC-1981) indicates that total (as opposed to individual) early fatality risk is very sensitive to site population distribution and that the mean number of early fatalities is determined by the average density of the exposed population. In the Sandia siting study, there was a difference of three orders of magnitude in the calculated mean number of early fatalities among 91 sites using the same source term, same emergency response, same wind rose, and same meteorological record. The only difference in this analysis was the use of a site-specific population distribution. On the bases of 1970 census data, the population density within five miles of the plant among the 91 sites examined had the following characteristics:

Median	- 40 people per square mile
90th Percentile	- 190 people per square mile
Maximum	- 790 people per square mile

Figure 1 shows the calculated number of early fatality results from this study.

Benchmark calculations were performed for the "Task Force on Interim Operation of Indian Point" (NRC-1980) to assess the impact of population density on offsite consequences. For these calculations, four high-density sites (Zion, Indian Point, Limerick, and Fermi), one average-density site (Palisades) and one low-density site (Diablo Canyon) were selected. A

standard plant was then located at each site, identical weather sequences and emergency responses were assumed, and wind-rose weighted 1970 population distributions for each site were utilized. This analyses allowed the comparison of the calculated total early fatality risk for each site. Figure 2 summarizes the results of this comparison in the form of complementary cumulative distribution functions (CCDFs) for total early fatality risk. This figure shows that the three sites with the highest population density have similar risk profiles and are substantially above the risk curves for the "average" site and for low-population-density sites. On Table 1 these curves have been reduced to single values showing the expected consequences (number of early fatalities per year). From this table, it is observed that the total early fatality risk is an order of magnitude greater for the three highest population density plants than for the "average" plant and more than two orders of magnitude greater than for a low-population-density plant.

Table 1
Expected Annual Consequences

Site	Early Fatality Risk
Indian Point	6.1×10^{-3}
Zion	4.7×10^{-3}
Limerick	3.5×10^{-3}
Fermi	9.2×10^{-4}
Palisades	2.9×10^{-4}
Diablo Canyon	1.6×10^{-5}

Figure 3 shows the distribution of population densities within one-mile wide annular rings within five miles of each nuclear plant (population densities projected for the year 2000) (NRC-1979). This figure indicates that the population densities for the top 10 percent of nuclear plant sites exceed the median site population by approximately one order of magnitude. These results indicate that if all plants were to exactly meet the individual early fatality risk QHO then the top 10 percent of the plants would contribute the majority of the societal risk.

These results suggest that a number of the high-population-density sites may have an order of magnitude or larger societal risk than the median site. For plants at these sites, it is not clear that a risk metric that only considers individual risk is adequate. Consequently, the Committee may wish to consider whether societal risk should be considered in applying the safety goals on a plant-specific, site-specific basis.

Strawman Approach

One possible approach for addressing societal risk impacts for high-population-density sites that would not result in the need for revisiting the definition of the safety goals and could be implemented using the approach delineated in draft Regulatory Guide DG-1061 (NRC-1997) is discussed below.

For the very high-population-density sites, the QHOs for individual early fatality risk and individual latent cancer fatality risk could be reduced from their nominal values by a "population density reduction factor" to compensate for the increased societal risks associated with the high-population-density sites. On the basis of the results presented above, a reduction of approximately a factor of 10 in the individual early fatality QHO would be required for the highest population density sites to assure that the societal early fatality risk for these sites would not exceed the societal risk for an "average" site. The reduced QHO could then also be used to calculate a site-specific LERF. However, for these sites, it seems prudent to require a full Level 2/Level 3 probabilistic risk assessment and direct comparison with the (reduced) QHOs because of the large societal risk potential. The use of the simplified containment analyses procedures contained in Appendix B to draft Regulatory Guide DG-1061 and use of a LERF as surrogate for the QHOs may not be appropriate. The core damage frequency (CDF) could probably remain at 1×10^{-4} since this value for the CDF has been demonstrated in NUREG-1150 to result in an individual early fatality risk that is substantially below the safety goal QHOs.

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- cc: J. Larkins, ACRS
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 ACRS Senior Engineers

Figure 1

Histogram of Mean Early Fatalities for 91 Sites, Conditional on an SST1 release. Assumptions: 1120 MWe reactor, a representative meteorological record, and Summary Evacuation.

FIGURE 2 EARLY FATALITY RISK FOR DIFFERENT SITES

NOTE: THERE ARE LARGE UNCERTAINTIES WITH THE ABSOLUTE VALUES PRESENTED IN THIS FIGURE

- ASSUMPTIONS:
- 1) SURRY DESIGN.
 - 2) I.P. UNIT 3 POWER LEVEL (3025 MW).
 - 3) WITHIN 10 MILES - ENTIRE CLOUD EXPOSURE + 4 HOURS GROUND EXPOSURE
NO SHIELDING
BEYOND 10 MILES - ENTIRE CLOUD EXPOSURE + 7 DAY GROUND EXPOSURE
SHIELDING BASED ON NORMAL ACTIVITY.
 - 4) WIND ROSE WEIGHTED 1970 CENSUS POPULATION DISTRIBUTION.
 - 5) IDENTICAL 91 WEATHER SEQUENCES FOR ALL SITES.

RESIDENTS PER SQUARE MILE

FIGURE 3

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(See instructions on the reverse)

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10 SUPPLEMENTARY NOTES

11 ABSTRACT (200 words or less)

This compilation contains 67 ACRS reports submitted to the Commission, or to the Executive Director for Operations, during calendar year 1997. It also includes a report to the Congress on the NRC Safety Research Program. All reports have been made available to the public through the NRC Public Document Room, the U S Library of Congress, and the Internet at <http://www.nrc.gov/ACRSACNW>. The reports are categorized by the most appropriate generic subject area and by chronological order within the subject area.

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